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Generation IV Roadmap

R&D Scope Report for Gas-Cooled Reactor Systems

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MEMBERS OF THE GAS-COOLED REACTOR SYSTEM CONCEPTS TECHNOLOGY WORKING GROUP

Franck Carré	Co-Chair	Atomic Energy Commission–France
Philip Hildebrandt	Co-Chair	Engineering, Management, and Technology, Inc., United States
Finis Southworth	Technical Director	Idaho National Engineering and Environmental Laboratory, United States
Timothy Abram		British Nuclear Fuels Limited, United Kingdom
Sydney Ball		Oak Ridge National Laboratory, United States
Bernard Ballot		Framatome – Advanced Nuclear Programs, France
Phillip Finck		Argonne National Laboratory, United States
Konstantin Foskolos		Paul Scherrer Institute, Switzerland
Kosaku Fukuda		International Atomic Energy Agency, United Nations
Dominique Greneche		COGEMA, France
Andrew C. Kadak		Massachusetts Institute of Technology, United States
Shin Whan Kim		Korea Power Engineering Company, Korea
Werner Von Lensa		Forschungszentrum Juelich, European Commission
Masuro Ogawa		Japanese Atomic Energy Research Institute, Japan
Arkal Shenoy		General Atomics, United States

OTHER CONTRIBUTORS

Hussein Khalil	RIT Representative	Argonne National Laboratory, United States
William Naughton	GRNS Representative	Exelon, United States
Jacques Royen	Gas TWG Secretariat	Nuclear Energy Agency, Organisation for Economic Co-operation and Development
John M. Ryskamp	RIT Representative	Idaho National Engineering and Environmental Laboratory, United States
Bob Seidel		Argonne National Laboratory, United States
Steven Sorrell	DOE Representative	Department of Energy–Idaho Operations Office, United States

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ABSTRACT

The summary report on gas-cooled Generation IV concepts (Gas-TWG, 2001) evaluated the potential to achieve Generation IV goals for 21 different gas-cooled nuclear reactor system concepts. In that report, 19 concepts were aggregated into four concept sets for detailed evaluation, with each set being represented by a reference concept. These four reference concepts, which satisfied this initial screening for potential, were a pebble bed modular reactor system (PBR, G1), a prismatic modular reactor system (PMR, G2), a very high-temperature gas-cooled reactor system (VHTR, G3), and a fast neutron spectrum gas-cooled reactor system (GFR, G5). An additional concept, a thermal-spectrum high-conversion gas-cooled reactor system (generic closed cycle-GCC, G4) was added during subsequent quantitative scoring evaluations as a variation to the PMR. This report presents a summary description of the research and development scope for the four concept sets retained for further evaluation by the Gas-cooled Reactor Systems Technical Working Group.

EXECUTIVE SUMMARY

The objective of the second stage of the Generation IV Roadmap project is to further evaluate and determine the research and development (R&D) scope for the nuclear energy system concepts that have the potential to fulfill the Generation IV goals. Generation IV concepts are to have the potential for substantive improvements upon third generation light water reactor systems (LWRs) in the goal areas of sustainability, safety and reliability, and economic performance. *Description of Candidate Gas-Cooled Reactor Systems Report* (Gas-TWG 2001; available as Document 16 on this CD) is a summary report of the candidate gas-cooled reactor (GCR) system concepts that have potential to fulfill Generation IV goals.

In this report, the Gas-Cooled Reactor Systems Technical Working Group (Gas-TWG) has further developed the evaluation of the candidate concepts and identified a number of R&D needs required to support the design and application of similar or related reactor system concepts. The Gas-TWG members are broadly experienced in the design, construction, and operation of reactors, in particular GCRs, and represent several countries and international organizations.

The GCR systems described and evaluated herein are derived from the 21 summary concept descriptions provided (a) in public response to the U.S. Department of Energy request for information, and (b) by members of the Gas-TWG. The latter descriptions were prepared to complement the public response to ensure that the broadest range of candidate systems were considered based on the collective knowledge of the Gas-TWG members, and to include review of selected reports prepared by other agencies worldwide.

Four Reference Gas-Cooled Concepts

The concepts considered are grouped into concept sets, representing the common capabilities and attributes among the concepts. The concept sets provide the ability to evaluate the aggregate characteristics of the concepts in the set and to identify the common technical needs for subsequent consideration as potential R&D activities in the Generation IV Technology Roadmap. A reference reactor system has been chosen to represent each concept set, as follows:

- Modular Pebble Bed Reactor Systems (PBR open cycle, G1)
- Prismatic Fuel Modular Reactor Systems (PMR open cycle, G2)
- Very-high-temperature Reactor Systems (VHTR open cycle, G3)
- Gas-Cooled Fast Reactor Systems (GFR closed cycle, G5).

As part of quantitative scoring of concepts using the methodology prescribed by the Evaluation Methodology Group, an additional reference concept representative of a thermal spectrum, prismatic fuel modular reactor with a closed uranium/thorium fuel cycle was characterized (Generic Gas closed cycle, G4). This is a variation on G2.

The four reference concepts constitute a comprehensive family of nuclear energy systems that envelope a wide range of applications (e.g., electrical power generation, high-temperature process heat, waste destruction, and high sustainability via fissile production) with strong synergies in R&D needed for their commercial deployment. The pebble bed technology and the prismatic fuel technology that are the bases for two of the concept sets lead to industrial projects that are concluded to satisfactorily meet Generation IV goals and are judged to be deployable around 2010 to 2015. The other concept sets illustrate the potential of gas-cooled systems for enhanced performance, both toward higher temperatures to achieve very high conversion efficiencies and enhance alternative fuel production (e.g., hydrogen), and towards enhanced sustainability. This enhanced sustainability can be achieved through efficient and flexible use of available fissile and fertile nuclear fuels, efficient burning of long-lived radioactive waste, and enhanced intrinsic and extrinsic proliferation resistance.

Evaluation of the reference concepts was performed using the screening-for-potential criteria and the quantitative scoring methodology prescribed by the Evaluation Methodology Group with the concurrence of the TWGs. The screening-for-potential criteria provide a judgmental framework in which to evaluate the concept sets and have required selected adjustments for application to the GCR systems described herein. The quantitative methodology provides a reasonable basis for thorough evaluation and comparison of individual concepts, but in many respects the process is limited. The reactor system concepts under consideration cover a wide range of technologies, potential capabilities, and technical maturity and have different potential applications along the roadmap. It was found that the immaturity of the candidate concepts and associated extent of uncertainties often required qualitative judgment to perform the quantitative evaluations.

Results of Evaluations

Fulfill Generation IV Goals

Each of the reference reactor system concepts has the potential to fulfill the goals for a Generation IV nuclear energy system.

Robust Safety Performance

A technical basis exists to conclude that many of the GCR systems would not incur fuel damage under accident conditions and could be certified to meet protective action guidelines at the site boundary. As a consequence, a high pressure, conventional-type containment is considered unnecessary, and the size of emergency planning zone (EPZ) could be greatly reduced compared to the reference advanced LWRs, with the EPZ anticipated to extend only to the site boundary.

This is achieved by using refractory-coated fuel particles (e.g., using SiC coatings) and a limited size core at low power density. The particle fuel consists of a spherical kernel of fissile or fertile fuel material encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion-resistant pressure vessel and a barrier that is essentially impermeable to the release of gaseous and metallic fission products. This capability has been demonstrated at temperatures in excess of those that are predicted to be achieved under worst-case accident conditions.

Broad Spectrum of Applications

The GCR systems provide thermodynamic conditions that support potential applications ranging from electrical power generation to alternate fuel production (e.g., hydrogen) that make substantial in-roads in reducing production of carbon gases released to the atmosphere. These systems achieve high thermodynamic efficiencies and have application potential well beyond the reference advanced LWRs. For example, the pebble bed and prismatic fuel concept sets achieve outlet temperatures of about 850°C, leading to thermal efficiencies approaching 48% for a direct Brayton cycle gas turbine. At these temperatures, hydrogen production is practical via steam reforming or iodine-sulfur (IS) processes. In the case of the VHTR system concept sets, temperatures above 900°C and potentially approaching 1200°C may be achieved, with thermal efficiencies approaching 60%. Additional potential high-temperature applications, such as coal gasification and glass manufacture, as well as efficient production of hydrogen, could be possible.

Flexibility to Accommodate Multiple Fuel Cycles

The operating characteristics of the GCR systems accommodate use of a wide range of fuel cycles without changing the basic system design. The applicable fuel cycles range from low-enriched uranium to thorium-uranium to plutonium alone. Prismatic block GCRs can also utilize the discharged and separated transuranic (TRU) waste from other reactors. The expected high burnup capability of the tri-isotopic (TRISO) fuel particle allows for the fission of over 90% of the fissile plutonium in the TRU in a single irradiation cycle. Effectively, the prismatic block reactor can destroy a significant part of its own waste.

The thermal and epi-thermal GCR concepts provide the capability to achieve much higher burnups compared to the reference advanced LWRs, with the attendant capability to burn most of the minor actinides, thereby reducing the waste heat load and radio-toxicity. Although capable of improved sustainability compared to an open-cycle concept, the thermal closed cycle GCR (G4) is limited to about a factor of two improvements in fissionable resource utilization. A considerably greater resource utilization is achievable with the gas-cooled fast spectrum reactor system.

High Sustainability

To achieve even greater sustainability improvements beyond the thermal GCR, a fast neutron spectrum GCR system can be used. This reactor system concept has a closed fuel cycle using high conversion or breeding of fissile materials. A breeding capability around unity may be of interest if a synergistic fuel cycle with LWRs is desired. The fast neutron spectrum reactor system provides a breeding capability that affords burning of about 70% of the energy of the natural uranium compared to only a few percent for thermal systems. The fast neutron spectrum reactor can use depleted uranium as the make-up fuel, thus precluding further mining of natural uranium.

R&D Scope

The R&D needs for GCR systems range from investigation of selected technical uncertainties (e.g., fuel microsphere containment integrity for more mature reactor systems) to fundamental R&D (e.g., for fuel design and materials for very-high-temperature reactor system and fast neutron spectrum reactor system applications).

Recommendations

These four GCR concepts constitute a progressive group of reactor systems with complementary R&D activities. The concepts rely on a common R&D pathway composed of basic needs for potential near-term concepts (i.e., PMR and PBR) and more ambitious objectives for the advanced concepts with the potential for extended capabilities (i.e., VHTR as a higher-temperature heat source for more efficient electricity generation and alternate fuel production, and GFR to achieve greatly improved sustainability). The following recommendations apply:

1. The R&D activities for the PBR and PMR are important precursors for the development of VHTR and GFR. The Technical Roadmap should describe this relationship and provide for periodic review of the status and success of the development and application of these nearer-term concepts to confirm that the ongoing R&D scope is adequately comprehensive. Further, as the practical aspects of funding priorities and cooperative development are realized, particular priority should be

given to those viability or performance issues that are most broadly applicable across the spectrum of GCR systems.

2. The VHTR concept should not be approached as a reactor system with a specific operating temperature and particular energy conversion process. Rather, R&D activities should be directed toward achieving the capability for increased temperatures at several points over a range from 950 to 1100°C, since the materials that may succeed for fuel coating and plant equipment could be expected to change markedly over this range. This approach would not only provide the potential for higher-temperature applications, but also provide materials with additional margins for use at temperatures applicable to PBR and PMR.

   Similarly, for energy conversion development, various candidate applications (e.g., hydrogen production and electrical power generation) and more than one way to achieve an application should be considered (e.g., for hydrogen, steam reforming and thermochemical processes).

3. Where available and practical, the R&D activities should be aligned to current and ongoing development activities that are not currently associated with the Generation IV program. For example, the High-Temperature Engineering Test Reactor (HTTR) in Japan provides an opportunity for complementary development. In this case, an initial research activity could be to review the current scope of HTTR, its status and planned additional scope against the overall R&D activities for a VHTR concept with the intent of reaching agreement on sharing information and cooperatively using this facility for Generation IV work.

4. A particular weakness in the concept evaluations was estimating the expected costs for building and operating the reactor system. This becomes even less certain when attempting to estimate whether and under what conditions the reactor system concept could be expected to be competitive for production of alternate fuels. Economic studies should have an early and generic priority (e.g., to better establish the expected market place for alternate fuels) to address conceptual issues such as the potential economic tradeoffs on small, modular reactor concepts versus the more conventional wisdom regarding large facility economy of scale.

   Further, the utilization of certain of these reactor system capabilities would be expected to be determined not by the marketplace, but by forward-looking governmental policies regarding carbon-based energy resource utilization, nuclear fissile and fertile resource utilization, non-proliferation, and nuclear waste management. R&D activities and studies, the results of which assist in shaping government policies for utilization of these capabilities, should be a high priority in the Generation IV Roadmap scope.

5. A weakness in both the concept evaluations and the description of R&D scope is the approach to addressing proliferation resistance and physical protection. Early studies should be directed at defining the conceptual standards that should be used in characterizing the threats, evaluating the reactor system vulnerabilities, and designing reactor systems that have improved capabilities for these considerations. In addition to the obvious desire to ensure the safety of the general population, these standards could be expected to affect conclusions regarding the specific and comparative economic viability of reactor systems.

ACRONYMS

AOO	Anticipated Operational Occurrences
ALWR	Advanced Light-Water Reactor
APBR	Annular Pebble Bed Reactor
AVR	ARBEITSGEMEINSCHAFT VERSUCHSREAKTOR (in Germany)
BDBA	beyond design basis accident
C/C	Carbon/Carbon
CERCER	ceramics fuel
CERMET	ceramic/metal
CHP	combined heat and power
CRD	control rod drives
DBA	design basis accident
DOE	Department of Energy
EG	electricity generation
EPA	Environmental Protection Agency
EPZ	emergency planning zone
FHM	fuel handling machine
GA	General Atomics
Gas-TWG	Gas-Cooled Reactor Systems Technical Working Group
GCR	gas-cooled reactor
GFR	gas-cooled fast reactor
GTG	Gas Turbine Generator
GT-MHR	Gas-Turbine Modular Helium Reactor
GWe	gigawatt-electric
H_2-MHR	hydrogen-producing modular helium reactor
HAT	high-temperature, accelerator-driven transmuter
HEU	high enriched uranium
HFR	High-Flux Reactor
HIT	high-temperature, inherent-safe transmuter
HPTF	high-pressure test facility
HTTR	high-temperature engineering test reactor
HX	heat exchanger
IC	intercooler
IFMU	in-core flux mapping unit
IHX	intermediate heat exchanger
IPyC	inner pyrolytic carbon
IS	Iodine-Sulfur
JAERI	Japan Atomic Energy Research Institute
LEU	low enriched uranium
LOCA	loss of coolant accident
LWR	light water reactor
MEDUL	multiple-recirculating feeding system
METMET	composite metal
MHR	modular helium reactor
MHTGR	Modular High-Temperature Gas-Cooled Reactor
MOX	mixed oxide
MW	megawatt
MWe	megawatt-electric

MWt	megawatt-thermal
NCS	neutron control system
NDE	nondestructive examination
NDTT	nil ductivity transition temperature
NERI	Nuclear Energy Research Initiative
NHSS	Nuclear Heat Supply System
NPH	Nuclear Process Heat
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
OTTO	once-through-then-out
PBR	Module Pebble Bed Reactor
PC	precooler
PBMR	Pebble Bed Modular Reactor
PCIV	prestressed cast-iron vessels
PCM	Phase Change Material
PCU	power conversion unit
peu-a-peu	little-by-little
PIE	postirradiation examinations
PMR	Prismatic Fuel Modular Reactor
PNP	Prototype Nuclear Process Heat
PRA	Probabilistic Risk Assessment
PWR	pressurized water reactor
QA	Quality Assurance
QC	Quality Control
R&D	research and development
RCCS	reactor cavity cooling system
RSE	Reactor Service Equipment
SCS	Shutdown Cooling System
SCS-HX	Shutdown Cooling System-Heat Exchanger
SLSV	shutdown loop shutoff valve
SNL	Sandia National Laboratory
THTR	Thorium High-Temperature Reactor
TINTE	Space Time Kinetics Code
TP	thermoplastic
TRISO	tri-isotopic
TRU	transuranic
TS	thermosetting
TWG	Technical Working Group
UCO	uranium oxycarbide
VHTR	very-high-temperature reactor
VLPC	vented, low-pressure containment
VSOP	very special old programs

R&D Scope Report for Gas-Cooled Reactor Systems

1. INTRODUCTION

The primary goal of the U.S. Department of Energy (DOE) Generation IV Roadmap Development initiative is to identify and evaluate advanced nuclear energy system concepts that offer significant advances toward meeting stringent performance goals of sustainability (e.g., resource utilization, waste minimization, environmental impact, and proliferation resistance), safety and performance, and economy. The goal of the Generation IV initiative is to identify and develop next generation nuclear energy systems that are deployable by 2030 and can meet energy needs through the 21st century. The Gas-Cooled Reactor Systems Technical Working Group (Gas-TWG) was formed in January 2001 as one of four technical working groups supporting the Generation IV Technology Roadmap.

During the preparation of this report, the Gas-TWG comprised eleven members from the United States and ten members representing England, France, Japan, Korea, the European Commission, the International Atomic Energy Agency and the Nuclear Energy Agency. The Nuclear Energy Agency provides the secretariat function for the Team. Team members are drawn from universities, national laboratories, government agencies, and industry, and are broadly experienced in the design, construction, and operation of reactors, in particular, gas-cooled reactors (GCRs). This membership provides an international perspective and opinion of the potential for GCR systems to fulfill the goals for sustainability, safety and reliability, and economics for a Generation IV nuclear energy system.

Twenty-one high-temperature, GCR system concepts were contributed to the Gas-TWG, forming the starting point concepts for developing a research and development (R&D) roadmap. Most of these concepts, for purposes of identifying technical uncertainties or innovations to support Generation IV performance goals, were aggregated into the four nuclear energy system concept sets. A reference reactor system concept was chosen to represent each of these sets for further evaluation.

GCR systems have several fundamental characteristic features that distinguish them from other types of reactors and provide significant operational advantages. In particular, the fuel is in the form of small ceramic-coated particles capable of very high-temperature operation, the moderator is solid graphite, and the coolant is neutronically inert helium or carbon dioxide. One of the benefits of such a fuel arrangement is that GCR systems are able to accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design. This flexibility is due to an uncoupling between the parameters of cooling geometry and the parameters that characterize neutronic optimization (i.e., moderation ratio or heavy nuclide concentration and distribution). It is possible to modify the packing fraction of coated particles in the fuel within the graphite matrix without changing the dimensions of the fuel elements (number and diameter of cooling holes for prismatic block cores or pebble diameter for pebble bed cores). Other physical reasons favour the adaptability of GCR systems with regard to the fuel cycle in comparison with reactors using moderators in the liquid form, such as liquid water reactors (LWRs). One illustration is the void coefficient, which limits the plutonium content of pressurized water reactor (PWR) mixed oxide (MOX) fuels[a] and which is not a constraint for GCR systems. Also, a GCR core exhibits less parasitic capture in the moderator (the capture cross section of graphite is 100 times less than the one of water) and in internal structures.

a. If a total loss of water occurs in a PWR, the neutron spectrum becomes very fast due to the reduced moderation. In these conditions, neutron multiplication by plutonium isotopes increases significantly because of better neutron reproduction of plutonium isotopes in the fast range.

Finally, the fuels in GCR systems are able to reach very high burn-ups, which are far beyond the possibilities offered by other thermal reactors (except the particular case of molten salt reactors). This capability allows for essentially complete plutonium fission in a single burnup and minimizes the proliferation risk in the use of this fuel form.

2. CONCEPT DESCRIPTIONS AND FINAL SCREENING EVALUATIONS

The GCR systems described and evaluated herein are based on the 21 summary concept descriptions provided in public response to the DOE request for information and by members of the Gas-TWG. The latter descriptions were prepared to complement the public response to ensure that the broadest range of candidate systems were considered based on the collective knowledge of the Gas-TWG members, and to include review of selected reports prepared by other agencies, worldwide.

Nineteen of the 21 concepts considered are grouped into the following four concept sets, representing the common capabilities and attributes among the concepts:

- Modular Pebble Bed Reactor Systems (PBR open cycle, G1)
- Prismatic Fuel Modular Reactor Systems (PMR open cycle, G2)
- Very-high-temperature Reactor Systems (VHTR open cycle, G3)
- Gas-Cooled Fast Reactor Systems (GFR closed cycle, G5).

As part of quantitative scoring of concepts using the methodology prescribed by the Evaluation Methodology Group, an additional reference concept, representative of a thermal spectrum, prismatic fuel modular reactor with a closed uranium/thorium fuel cycle, was characterized (Generic Gas closed cycle, G4). This is a variation on G2.

The reference concepts provide the ability to evaluate the aggregate characteristics of the concepts in the set and to identify the common technical needs for subsequent consideration as potential R&D activities in the Generation IV Technology Roadmap. Each reference concept is described below

2.1 Pebble Bed Reactor Systems

Key design characteristics of both PBRs and PMRs (see Section 2.2) are the use of helium coolant, graphite moderator, and refractory tri-isotopic (TRISO)-coated particle fuel. The helium coolant is inert and remains single phase under all conditions; the graphite moderator has high strength and stability to high temperatures; and the TRISO-coated particle fuel retains fission products to high temperatures. The TRISO-coated particle fuel consists of a spherical kernel of fissile or fertile material, as appropriate for the application, encapsulated in multiple coating layers. The multiple coating layers form a miniature, highly corrosion-resistant pressure vessel and an essentially impermeable barrier to the release of gaseous and metallic fission products.

In the PBR concepts, the TRISO-coated microspheres are contained in a 6-cm ball configuration as the fuel form (i.e., the "pebble"). In the inner 5 cm, the coated particles are homogeneously distributed within a graphitic matrix surrounded by a 0.5-cm outer pyrocarbon shell, with the carbon particle coatings, matrix, and shell acting as moderator. There are two generic concepts for PBRs in terms of refueling. The most common is the online, multiple-recirculating feeding system (MEDUL) in which pebbles are continuously removed, controlled with regard to their burn-up and mechanical integrity, then transported back to the top of the reactor core if they have not yet reached the burn-up target. Fresh fuel is only added as needed to maintain criticality over the full range of operating conditions. The other types are the once-through-then-out (OTTO) concept, where the pebbles only perform one passage through the

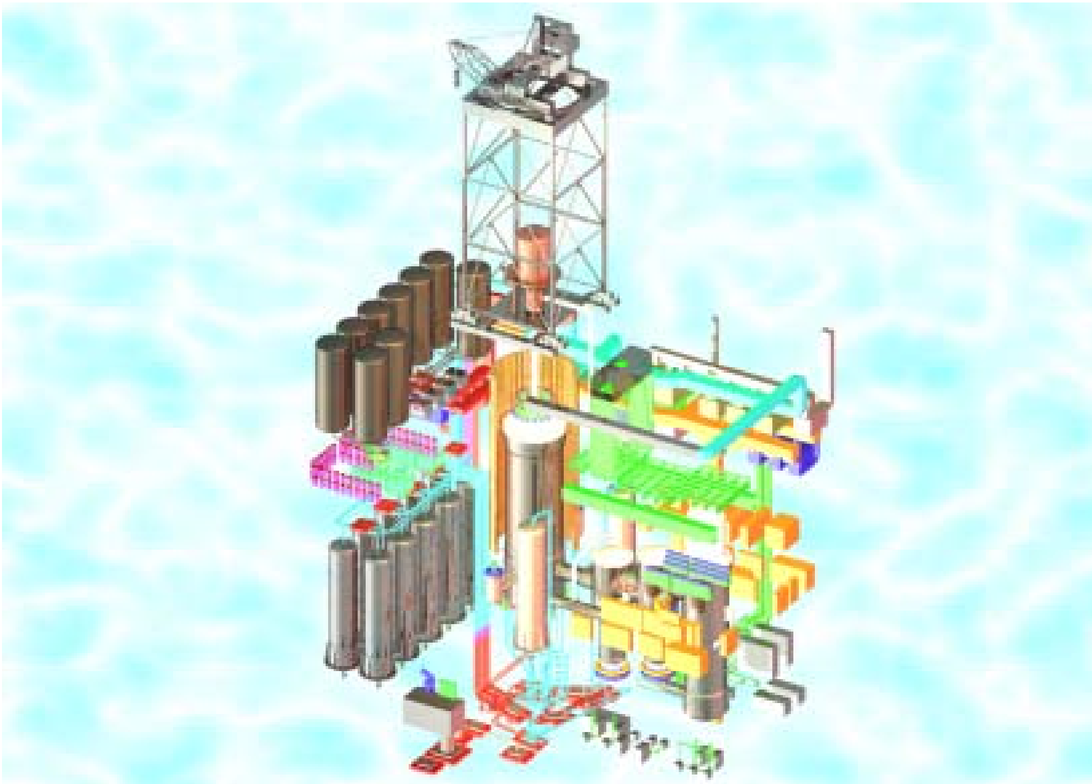

Figure 1. Pebble Bed Modular Reactor Power Plant cutaway, courtesy of PBMR, Pty.

core, and the little-by-little (*PEU-A-PEU*) scheme, in which fuel is added to maintain criticality then completely discharged and replaced when the vessel is full. For the on-line refueling designs, the pebbles are circulated by gravity in the core, which is surrounded by a graphite reflector, and pneumatically transported in the fuel handling system. Normally, the graphite reflector is not easily exchangeable and has to resist high neutron fluences and thermal effects that accumulate during lifetime, although future designs contemplate replacement if required.

The PBR concepts use a thermal neutron spectrum and have the capability to maintain fuel integrity under all design basis accidents (DBA) with no reliance on active safety systems for short-term safety functions. Long-term shutdown systems are required for anticipated transients without scram events. Analyses have shown that the core cannot melt down. The refractory core, low power density, and low excess reactivity enable this design approach. PBRs also exhibit high efficiency with either a direct or indirect gas turbine power conversion system, with or without a bottoming cycle, using the relatively high exit temperature (about 500°C) helium from the turbine. The reference PBR is 250 MW thermal and 115 MW electrical. Other variations use intermediate heat exchangers to facilitate process heat applications or steam cycle power conversion while maintaining moisture isolation from the primary coolant circuit. Online refueling of the PBRs leads to low excess reactivity in the core while allowing very high reactor availability.

Table 1. PBR nominal full power operating parameters.[a]

Reactor Power	250 MWt
Core Inlet/Outlet Temperatures	412/850°C
Core Inlet/Outlet Pressures	7.8/7.65 MPa
Helium Mass Flow Rate	120 kg/s
Turbine Inlet/Outlet Temperatures	800/373°C (4 turbines)
Turbine Inlet/Outlet Pressures	7.75/1.90 MPa
Recuperator Hot Side Inlet/Outlet Temperatures	373/125°C
Recuperator Cold Side Inlet/Outlet Temperatures	104/330°C
Net Electrical Output	110 MWe
Net Plant Efficiency	44%

a. These values are preliminary based on an unoptimized design using only proven technology.

2.1.1 Strengths and Weaknesses of the PBR

The design intent of the PBR reference concept is to excel in achieving Generation IV safety and reliability goals, as well as providing a low capital investment modular generating station to best permit a renewal of the nuclear electrical option. The PBR also builds upon the PBMR near-term concept, which has focused on achieving good plant economics (busbar cost). Regarding sustainability, the PBR is generally comparable to Generation III reference LWR concepts. Given the relatively early deployability of the PBR, this is to be expected since uranium costs are about 1% of the cost of electricity for today's nuclear plants.

2.1.1.1 Sustainability.

2.1.1.1.1 Sustainability 1—The PBR has high thermal conversion efficiency and high fuel burnup, made possible by the use of TRISO-coated particle fuel. This is offset by the PBR reference high enrichment (8 w/o), leading to uranium utilization comparable to the Generation III reference.

The passive safety characteristics of the PBR would result in no requirement for emergency planning for evacuations outside a small exclusion area zone around the reactor plant. As a consequence, the PBR would need much less land space than a comparably-sized LWR, and close-in siting of plants to points of energy use may be possible. Close-in siting of energy generation plants reduces both land and electricity distribution resource requirements and makes other process heat applications more attractive.

The thermal discharge (waste heat) from the PBR is one-half that for LWRs per unit of electricity produced (assuming 32% Advanced LWR efficiency vs. 42-48% PBR efficiency). If this waste heat were to be discharged using conventional power plant water heat rejection systems, the PBR would require one-half as much water coolant per unit of electricity produced. Alternatively, because of this, the PBR waste heat could be rejected directly to the atmosphere using air-cooled heat rejection systems such that no water coolant resources are needed. Because of this capability, the use of the PBR in arid regions is practical.

2.1.1.1.2 Sustainability 2: Waste Management—The PBR produces less heavy metal radioactive waste than the Advanced Light-Water Reactor (ALWR) and may be a superior fuel form for disposal in a geologic repository. Because of the higher enrichment and higher burnup of PBR fuel, compared to LWRs, the mass of heavy metal will be significantly reduced. Depending upon the direct disposal capability of the fuel, the volume is expected to be comparable. In terms of long-term heat output and radiotoxicity, the PBR is expected to be comparable to LWRs.

2.1.1.1.3 Sustainability 3: Proliferation Resistance and Safeguards—The PBR has high proliferation resistance and has been designed to satisfy international safeguard requirements. The PBR's high proliferation resistance is primarily due to the refractory-coated fuel form and the reactor's characteristically high burnup.

2.1.1.2 Safety and Reliability. The PBR excels in the safety criteria. Safety is achieved through a combination of inherent safety characteristics and design selections that take advantage of the passive safety characteristics. These characteristics and design selections include:

1. Helium coolant, which is single-phase, noncondensable, inert, and has no reactivity effects. With helium as the coolant, the cost of the gas makes a low leakage rate mandatory.

2. Graphite core, which provides high heat capacity, slow thermal response, and structural stability at very high temperatures. Power density is about 10 times lower than in an LWR.

3. Finely divided refractory coated particle fuel, which retains fission products at temperatures much higher than normal operation and postulated accident conditions, and has a statistical behavior.

4. Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures.

5. Inherently, very low excess reactivity as a result of continuous on-line refueling.

6. Limited total core power allowing ultimate heat sink capability by conduction and radiation while incurring no fuel damage.

For passive removal of decay heat, the core power density and the size have been designed such that the decay heat can be removed by heat conduction, thermal radiation, and natural convection without exceeding the fuel particle accident temperature design goal. Core decay heat is conducted to the pressure vessel and transferred by radiation and convection from the vessel to the natural circulation reactor cavity cooling system (RCCS). The RCCS provides an independent passive means for the removal of core decay heat in the event the two active, diverse heat removal systems—the power conversion system and a shutdown cooling system—are not available. Even if the RCCS is assumed to fail, passive heat conduction from the core, thermal radiation and convection from the vessel, and conduction into the silo walls provides the ultimate heat sink, sufficient to prevent core damage.

The plant design allows meeting the Environmental Protection Agency (EPA) protective action guideline limits at the fence boundary. In addition, the use of inert coolant and a simplified plant design provides extremely low routine and accident worker exposure. Exposures below the man-rem level are expected to be achievable with an uncertainty remaining for the amount of exposure needed for periodic maintenance of the gas turbine. In addition, the low power density, high thermal inertia core provides many days of thermal response time even to beyond design basis accidents (BDBA). The single-phase coolant and natural plant safety simplify analysis of accident conditions. Under these accident conditions, fuel does not melt and does not achieve temperatures that might be expected to degrade fission product retention.

2.1.1.3 Economics. Advances in the design of GCRs have been in the direction of more compact cores, made possible by the use of enriched uranium fuel and high operating temperatures. The pressure vessels can thus be smaller, which permits an increase in the design pressure (facilitating heat removal) at a reasonable cost.

The PBR has generating costs comparable to the reference LWRs. The construction cost is also comparable to LWRs (per GWe). The higher fuel fabrication cost is offset by lower operating staff cost. The economic risk is significantly lower because of the relatively small modular size of the PBR (110 MWe). The use of graphite-coated enriched fuel permits high specific power and high burnup, significantly reducing power costs. Fertile species (U238, Th232) can be used in order to attempt thermal breeding (Pu239, U233) as a contribution to lower power costs. The high temperatures possible in GCRs open up the possibility of using the coolant gas in a closed-cycle gas turbine to produce electricity without the necessity for generating steam.

2.1.1.4 Summary. The PBR excels in achieving the safety goal, is quite strong in the economics goals (especially economic risk), and provides sustainability performance similar to or slightly better than Generation III water reactors. Because the development needs are relatively low for the PBR, it is deployable early in the next thirty-year period.

2.2 Prismatic Fuel Modular Reactor Systems

Prismatic Fuel Modular Reactor (PMR) systems have the same key design characteristics as PBRs and use the same TRISO-coated particles except that they are shaped into different configurations. For the prismatic designs, TRISO-coated particles are mixed with a matrix and formed into cylindrical fuel compacts approximately 13 mm in diameter and 51 mm long. The fuel compacts are loaded into fuel channels in hexagonal graphite fuel elements measuring 793 mm long by 360 mm across flats. One hundred and two (102) columns of the hexagonal fuel elements are stacked 10 elements high to form an annular core. Reflector graphite blocks are provided inside and outside of the active core.

The PMR system uses a thermal neutron spectrum and is designed to maintain fuel integrity under all DBAs with minimal active safety system requirements. Batch refueling requires periodic refueling shutdowns, but the fuel cycle flexibility is appreciable. High-burnup, low-enriched uranium (LEU, more than 5%), once-through fuel cycles are the reference approach. The high thermal efficiency of the systems leads to better than current generation fuel utilization. The reference PMR power level is 600 MW thermal and 286 MW electrical. Combinations of LEU, high-enriched uranium (HEU), plutonium recycle, thorium-uranium, and excess weapons material burning are examples of the fuel cycle flexibility exhibited by the PMRs. PMRs can also utilize the discharged and separated transuranic (TRU) waste from thermal reactors, and efficiently utilize the fissionable material, primarily plutonium, in this waste. The reference PMR concept has core exit coolant temperatures of about 850°C.

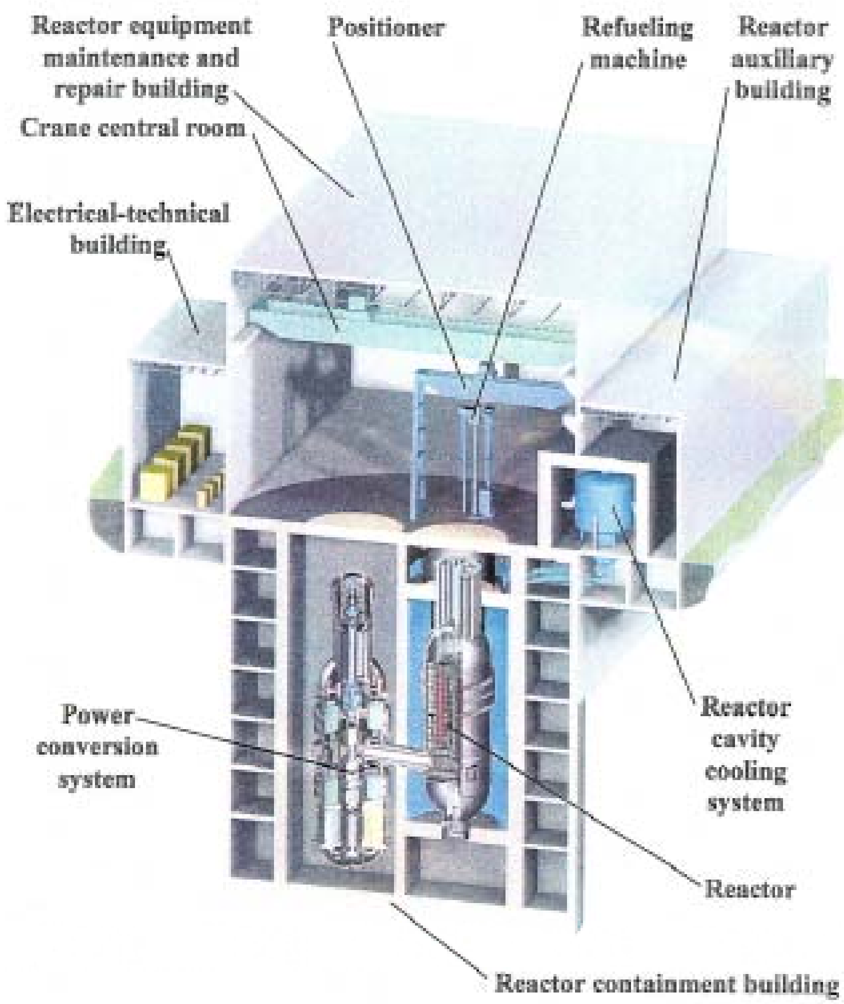

Figure 2. Gas-Turbine Modular Helium Reactor (GT-MHR) Power System.

Table 2. GT-MHR nominal full power operating parameters.

Reactor Power	600 MWt
Core Inlet/Outlet Temperatures	491/850°C
Core Inlet/Outlet Pressures	7.07/7.02 MPa
Helium Mass Flow Rate	320 kg/s
Turbine Inlet/Outlet Temperatures	848/511°C
Turbine Inlet/Outlet Pressures	7.01/2.64 MPa
Recuperator Hot Side Inlet/Outlet Temperatures	511/125°C
Recuperator Cold Side Inlet/Outlet Temperatures	105/491°C
Net Electrical Output	286 MWe
Net Plant Efficiency	48%

A variation of the PMR is the same reactor but with a closed, high-conversion fuel cycle (see Section 2.5 for a discussion of GCR fuel cycle flexibility). This concept was scored separately because it uses the closed (thermal spectrum) fuel cycle, but it adds no clarity to carry the concept as uniquely separate. In the near term (twenty years or more), uranium availability makes recycle of fuel economically undesirable. If uranium reserves later prove to be a concern, then the same reactor can utilize different fuel cycles that stretch available reserves. Similarly, waste volume and mass can be reduced by the recycle approach, when it is economically desirable to do so.

2.2.1 Strengths and Weaknesses of the PMR

The specific design intent of the PMR reference concept is to be the benchmark in achieving Generation IV safety and reliability goals. The PMR also builds on the Gas-Turbine Modular Helium Reactor (GT-MHR) near-term concept, which has focused on achieving excellent plant economics (busbar cost). Regarding sustainability, the PMR is generally comparable to Gen III reference LWR concepts. Given the relatively early deployability of the PMR, this is to be expected since uranium costs are about 1% of the cost of electricity for today's nuclear plants.

2.2.1.1 Sustainability.

2.2.1.1.1 Sustainability 1—The PMR has high thermal conversion efficiency and high fuel burnup, made possible by the use of TRISO-coated particle fuel. This is offset by the present reference high enrichment, leading to uranium utilization comparable to the Gen III reference.

Regarding the use of land resources, the passive safety characteristics of the PMR would result in no requirement for emergency planning for evacuations outside a small exclusion area zone around the reactor plant. As a consequence, the PMR would need much less land space than a comparably-sized LWR, making it possible to build the plants close to the points of energy use. Close-in siting of energy generation plants reduces both land and electricity distribution resource requirements and makes other process heat applications more attractive.

The thermal discharge (waste heat) from the PMR is one-half that for LWRs per unit of electricity produced (assuming 32% ALWR efficiency versus 48% PMR efficiency). If this waste heat were to be discharged using conventional power plant water heat rejection systems, the PMR would require one-half as much water coolant per unit of electricity produced. Alternatively, because of this, the PMR waste heat could be rejected directly to the atmosphere using air-cooled heat rejection systems such that no water coolant resources are needed. Because of this capability, the use of the PMR in arid regions is practical.

2.2.1.1.2 Sustainability 2: Waste Management—The PMR produces less heavy metal radioactive waste than the ALWR and may be a superior fuel form for disposal in a geologic repository. Because of the higher enrichment and higher burnup of PMR fuel compared to LWRs, the mass of heavy metal will be significantly reduced. Depending on the direct disposal capability of the fuel, the volume is expected to be comparable. In terms of long-term heat output and radiotoxicity, the PMR is expected to be comparable to LWRs.

2.2.1.1.3 Sustainability 3: Proliferation Resistance and Safeguards—The PMR has high proliferation resistance and has been designed to satisfy international safeguard requirements. The GT-MHR's high proliferation resistance is primarily due to the refractory-coated fuel form and the reactor's characteristically high burnup.

2.2.1.2 *Safety and Reliability.* The PMR excels in the safety criteria. Safety is achieved through a combination of inherent safety characteristics and design selections that take advantage of the passive safety characteristics. These characteristics and design selections include:

1. Helium coolant, which is single-phase, noncondensable, inert, and has no reactivity effects. With helium as coolant, the cost of the gas makes a low leakage rate mandatory.

2. Graphite core, which provides high heat capacity, slow thermal response, and structural stability at very high temperatures. Power density is about 10 times lower than in an LWR.

3. Refractory-coated particle fuel, which retains fission products at temperatures much higher than normal operation and postulated accident conditions, and has a statistical behavior.

4. Negative temperature coefficient of reactivity, which inherently shuts down the core above normal operating temperatures.

5. An annular, low power density core in an uninsulated steel reactor vessel surrounded by a natural circulation RCCS.

6. Limited total core power allowing ultimate heat sink capability by conduction and radiation while incurring no fuel damage.

For passive removal of decay heat, the core power density and the annular core configuration have been designed such that the decay heat can be removed by heat conduction, thermal radiation, and natural convection without exceeding the fuel particle accident temperature design goal. Core decay heat is conducted to the pressure vessel and transferred by radiation and convection from the vessel to the natural circulation RCCS. The RCCS provides an independent passive means for the removal of core decay heat in the event the two active, diverse heat removal systems—the power conversion system and a shutdown cooling system—are not available. Even if the RCCS is assumed to fail, passive heat conduction from the core, thermal radiation and convection from the vessel, and conduction into the silo walls provides the ultimate heat sink, sufficient to prevent core damage.

The plant design allows meeting the EPA protective action guideline limits at the fence boundary. In addition, the use of inert coolant and a simplified plant design provides extremely low routine and accident worker exposure. Exposures below the man-rem level are expected to be achievable with an uncertainty remaining for the amount of exposure needed for periodic maintenance of the gas turbine. In addition, the low power density, high thermal inertia core provides many days of thermal response time to even BDBAs. The single-phase coolant and natural plant safety simplify analysis of accident conditions. Under these accident conditions, fuel does not melt and does not achieve temperatures that might be expected to degrade fission product retention.

2.2.1.3 *Economics.* Advances in the design of GCRs have been in the direction of more compact cores, made possible by the use of enriched uranium fuel and high operating temperatures. The pressure vessels can thus be smaller; this permits an increase in the design pressure (facilitating heat removal) at a reasonable cost.

The PMR has comparable generating cost to the reference LWRs. The construction cost is comparable to LWRs (per GWe). The higher fuel fabrication cost is offset by lower operating staff cost. The economic risk is significantly lower because of the relatively small modular size of the PMR (288 MWe). The use of graphite-coated enriched fuel permits high specific power and high burnup, significantly reducing power costs. Fertile species (U238, Th232) can be used to attempt thermal breeding

(Pu239, U233) as a contribution to lower power costs. The high temperatures possible in GCRs open up the possibility of using the coolant gas in a closed-cycle gas turbine to produce electricity without the necessity for generating steam.

2.2.1.4 Summary. Overall, the PMR excels in achieving Gen IV safety and reliability goals, is strong in the economic goals, and affords similar sustainability performance as Gen III reference reactors.

2.2.2 Strengths and Weaknesses of the Generic Closed Cycle GCR (G4)

The G4 concept has the same strengths and weaknesses as the PMR. The reactor system is identical with a fuel cycle that relies on single or multiple fissile material recycle. This provides a modest gain in resource utilization (about a factor of two).

2.3 Very-High-Temperature Reactor Systems

Very-high-temperature Reactors have average coolant outlet temperatures above 900°C or operational fuel temperatures above 1250°C. These concepts provide the potential for increased energy conversion efficiency and for high-temperature process heat applications, such as coal gasification or thermochemical hydrogen production. While the temperature of all the GCR concepts considered are high enough to support process heat applications, such as desalination or cogenerative processes as well as some thermochemical processes of interest to alternative fuel production, the VHTRs higher temperatures open a broader and more efficient range.

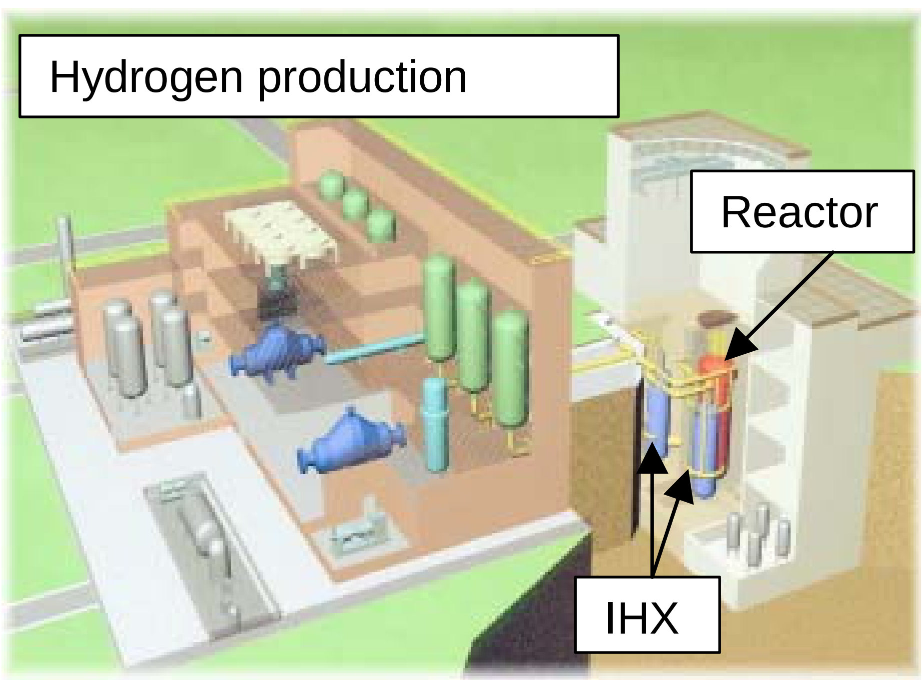

Figure 3. Schematic view of the VHTR with hydrogen production system.

These concepts require substantive improvements in the fuel design, especially in material properties. High-temperature alloys, fiber-reinforced ceramics or compound materials, and ZrC coatings of the fuel are promising candidates. The benefit of these developments is not restricted to dedicated VHTR applications but is valid for all kinds of high-temperature reactor applications irrespective of the core design. Thus, the VHTR concepts and applications can mainly be taken as an important direction for innovative, long-term, future R&D.

The reference concept has a block type core and is based on the GT-MHR connected to a steam reformer/steam generator unit in the primary circuit. It is an advanced, high-efficiency reactor system that can supply process heat to a broad spectrum of high temperature and can be used in energy-intensive, nonelectric processes. It can also be equipped with an intermediate heat exchanger (IHX), as is the case in the High Temperature Engineering Test Reactor (HTTR), to broaden the application spectrum. Pebble bed concepts are also applicable options for VHTRs.

All of the gas-cooled options can achieve temperatures suitable for the hydrogen production Gen IV mission. The VHTRs, however, are aimed at maximizing the efficiency of hydrogen production by stretching materials technology to new operating regimes. The VHTRs excel in achieving safety goals for Gen IV, may be excellent in economics for their hydrogen mission, and offer comparable sustainability to Gen III reference reactors.

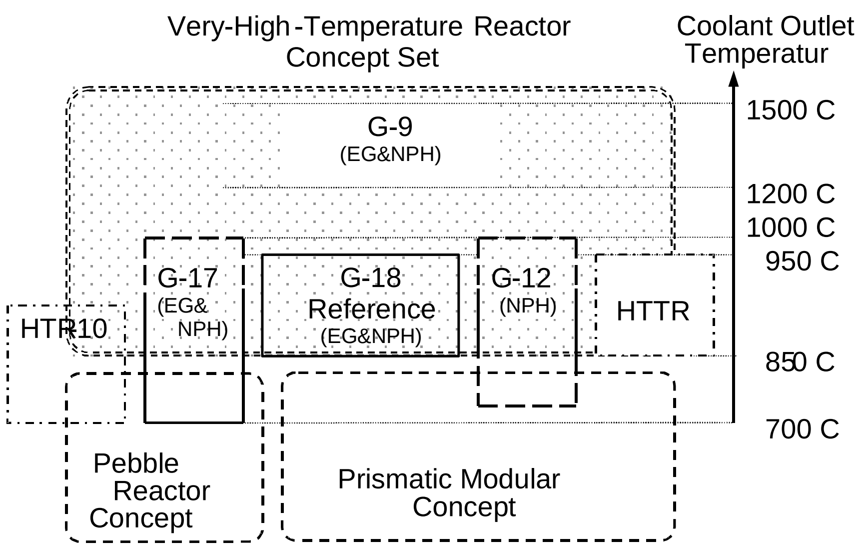

Figure 4. Relationship between VHTR concepts submitted and ongoing high-temperature reactors (HTRs).[b]

2.3.1 Strengths and Weaknesses of the VHTR

The VHTR has essentially the same strengths and weaknesses as the PMR (see Section 2.2) and is intended to stretch the PMR capabilities into more efficient power and process heat applications through higher coolant temperatures.

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b. G-9: Very-high-temperature Gas-Cooled Reactor (VHTR) for Electricity Generation (EG) and Transportation Fuel Production presented by General Atomics and Framatome; G-12: Modular Helium Reactor (MHR): nuclear heat source with block type core submitted by General Atomics; G-17: Annular Pebble Bed Reactor (APBR) for different applications proposed by FZ Juelich; G-18: Nuclear Process Heat (NPH).

2.4 Gas-Cooled Fast Reactor Systems

Gas-Cooled Fast Reactor (GFR) concepts offer a closed fuel cycle through high conversion or breeding of fissile materials. A breeding capability around unity may be of interest if the GFR is used in a symbiotic fuel cycle with LWRs. GFRs using a direct Brayton cycle have the potential to combine the advantages of high sustainability and economic competitiveness, while making nuclear energy benefit from the most efficient conversion technology available.

The reference concept is a 600 MWth/288 MWe, helium-cooled reactor system operating with an outlet temperature of about 850°C and using a direct Brayton cycle gas turbine. The thermal efficiency is estimated to approach 48%. There are several fuel design options including both the prismatic (with fuel particles or composite fuels) and fuel pins (with actinide compound/solid solution). The main design features of the concept are summarized in Table 3.

Table 3. Design features of the GFR concept.

Reactor Design Parameter	Conceptual Data
Power plant	600 MWth
Net efficiency (direct cycle helium)	48%
Coolant pressure	90 bar
Outlet coolant temperature	850°C (Helium, direct cycle)
Inlet coolant temperature	490°C (Helium, direct cycle)
Nominal flow and velocity	330 kg/s and 40 m/s
Core volume	10.9 m^3 (H/D ~1.7/2.9 m)
Core pressure drop	~ 0.4 bar
Volume fraction (%) Fuel/Gas/SiC	50/40/10%
Average power density	55 MW/m^3
Reference fuel compound	UPuC/SiC (50/50%) 17% Pu
Breeding/Burning performances	Self-Breeder
Maximum fuel temperature	1174°C (normal operation) < 1650°C (depressurization)
In core heavy nuclei inventory	30 tons
Fission rate (at %); Damage	~ 5 at%; 60 dpa
Fuel management	multi-recycling
Fuel residence time	3 × 829 efpd
Doppler effect (180°C–1200°C)	$-1540\ 10^{-5}$
Delayed neutron fraction	$356\ 10^{-5}$
Total He voidage effect	$+230\ 10^{-5}$
Average burn-up rate at EOL	~ 5% FIMA
Primary vessel diameter	< 7 m

2.4.1 Safety Design Provisions

For Loss of Fluid Accidents, GFRs are designed with natural convection and heat exchange, with the heat exchanger mounted at the top of the pressure vessel. For Loss of Coolant Accidents (LOCA), GFR designs facilitate long-term passive decay heat removal by conductive and radiative heat transfer across the core and the pressure vessel. Pressurized gas injection and natural convection at a back-up pressure of 5 to 15 bar (depending on the gas) is assured by the containment of the primary system.

Three barriers exist for the containment of fission products, with the containment of the primary system acting as a third barrier. A major challenge is to develop adequate fuel technologies and associated core designs (see Figure 5) and treatment processes to preserve most of the attractive safety features of thermal GCRs.

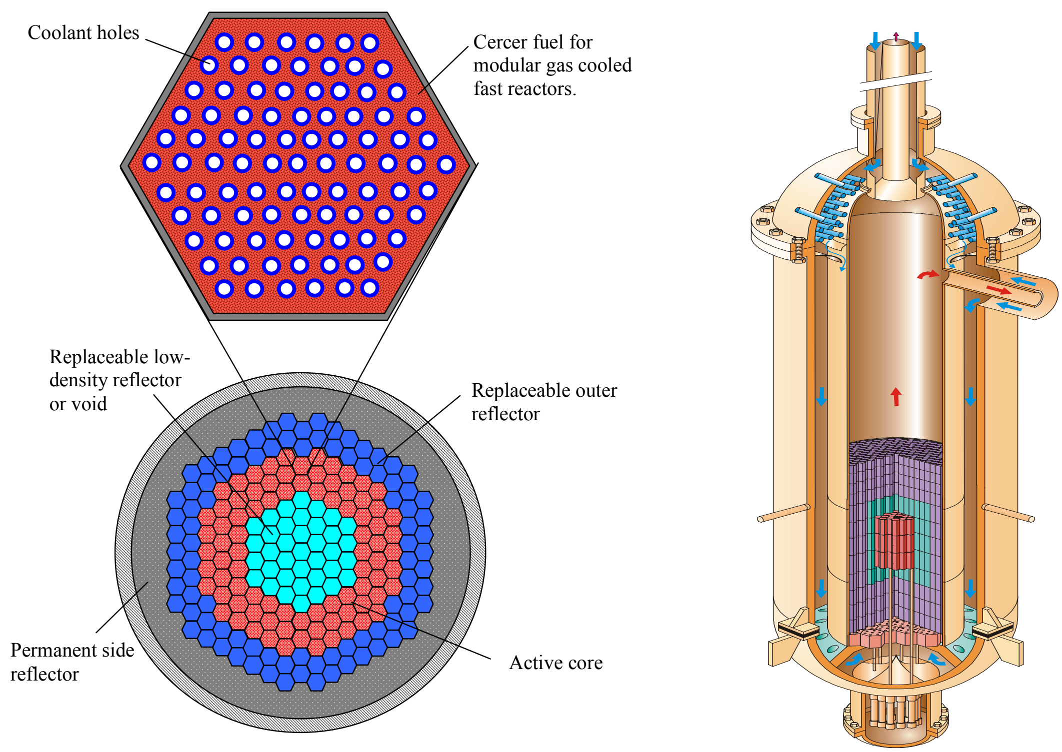


Figure 5. Schematic diagram of possible core layout with inner reflector for a modular, helium-cooled fast nuclear energy system with ceramics fuel (CERCER), or ceramics/metal (CERMET) or composite metal (METMET) as back-up solutions.

2.4.2 Strengths and Weaknesses of the GFR

2.4.2.1 Sustainability. The primary motivator for the GFR, is in the area of sustainability. The GFR utilizes multiple recycling of all actinides using pyrochemistry and remote fuel fabrication. The GFR, therefore, achieves both high uranium utilization and waste burning.

2.4.2.1.1 Sustainability–1: Effective Resource Utilization—

- The high thermal efficiency results in a more effective utilization of the nuclear fuels

- The hard neutron spectrum is intended to assure an efficient utilization of fertile nuclear fuels, including depleted uranium, while enabling high conversion and even breeding ratios

- Even though innovative fuel technology is needed, it is expected that benefits will be derived from the extensive experience gained on the particle fuel and composite transmutation fuel to achieve adequate particle or composite actinide fuel technologies

- It is compatible and flexible with a wide range of fuel cycles (Pu, U), (Pu, Th), HEU and Wpu.

2.4.2.1.2 Sustainability–2, Efficient Waste Management—

- The high thermal efficiency results in lower amounts of waste per unit of energy generated

- The fuel technology is compatible with closing fuel cycles of transuranic elements, thus making it possible to minimize the amount of long-lived radioactive materials in the waste

- Potentially, the integration of the fuel cycle on the production site minimizes the transportation of nuclear materials while restricting the materials leaving the site to various engineered waste forms containing only trace amounts of transuranic elements.

2.4.2.1.3 Sustainability–3, High Proliferation Resistance—

- The integration of the fuel cycle on the production site is a means to minimize proliferation risks

- High burn-ups and high initial transuranic content lead to fuel discharge compositions that are very unattractive for weapons use

- Safeguard measures and controls complement the above technical features to assure proliferation resistance.

2.4.2.2 Safety and Reliability. In achieving the safety and reliability goals, the GFR does not have the unique natural safety of the PMR (G2). The GFR does achieve passive safety, however, by the use of several design features. First, the total core power and the core power density have been limited to allow passive decay heat removal using natural convection. While these limitations decrease the potential breeding ratio and power plant economics, the balance results in a significant safety improvement over earlier versions of Gas-Cooled Fast Reactors (for example, the General Atomics GCFR of the 1970s).

- The fuel can operate in excess of 1650°C without loss of integrity, and the system is designed to maintain fuel temperatures below this level

- The helium coolant has negligible reactivity feedback effects (about half the delayed neutron fraction [~0.2 versus 0.35%]), and is not subject to any phase changes

- Negative reactivity feedback effects (an appreciable negative doppler coefficient [typically 1.5 to 2 10^{-5}/°C]) together with comfortable margins against design limits, will be used to the maximum extent possible to passively limit potential temperature or power transients.

Reactor systems are designed (e.g., ceramic materials, etc.) to facilitate rapid recovery and restart, even after postulating a simultaneous loss of coolant and loss of flow transient. A high-pressure (up to 15 bar) containment building is required to provide natural convection decay heat removal.

2.4.2.3 Economics. The GFR utilizes high-temperature helium coolant with a direct Brayton cycle power conversion to allow high efficiency. The cost of the power block is expected to be comparable to the PMR with the added costs of high-pressure containment and passive heat removal systems. The relatively small size of the power block allows for relatively low financial risk.

2.4.2.3.1 Economics–1, Clear Life-Cycle Cost Advantage—

- Helium coolant and ceramic fuel allow process temperatures as high as 850°C, potentially yielding high thermodynamic efficiencies for a broad spectrum of process heat applications.

- Detailed, DOE-sponsored, cost evaluations of gas-cooled, graphite-moderated, plants have shown, based on their expected performances, that they could be competitive with other systems for new electricity generation capacity and up-front capital costs, and produce similar power levels. By analogy, the unit costs for process heat are also anticipated to be competitive with fossil sources.

2.4.2.3.2 Economics–2, Financial Risk Comparable to Other Energy Projects—

- Initial process steam applications can efficiently operate with core outlet temperatures of 700°C, which is well within the operating experience of gas-cooled, graphite-moderated reactors that have been successfully operated at outlet temperatures as high as 950°C

- A modular design can be employed that permits incremental addition of generating capacity while minimizing the financial risk.

The GFRs, therefore, excel in achieving sustainability Gen IV goals, and offer the potential for moderate achievement of both safety and economic goals.

2.5 Gas-Cooled Reactor Fuel Cycle Flexibility

High-temperature GCR systems have several fundamental characteristics that distinguish them from and provide significant operational advantages over other types of reactors. In particular, the fuel is in the form of small, ceramic-coated particles capable of very high coolant temperature operation, the moderator is solid graphite, and the coolant is neutronically inert (e.g., helium or carbon dioxide). Note also that the coated particle fuel structure is expected to be capable of withstanding elevated temperatures.

One of the benefits of such a fuel arrangement is that the GCRs accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design. This flexibility is due to uncoupled cooling geometry and neutronic optimization (i.e., moderation ratio or heavy nuclide concentration and distribution). The solid moderator in GCRs also avoids the void coefficient, which limits the plutonium content of LWR MOX fuels.

High temperature GCR fuels are able to reach very high burn-ups, which are far beyond the possibilities offered by other thermal reactors (except the particular case of molten salt reactors). This capability allows for essentially complete plutonium fission in a single burnup and minimizes the proliferation risk in the use of this fuel form. Hence, the operating characteristics of the GCRs accommodate use of a wide range of fuel cycles without changing the basic reactor system design. The applicable fuel cycles range from LEU to thorium-uranium to plutonium alone.

3. TECHNOLOGY GAPS, REQUIRED R&D, AND R&D CHALLENGES

This section summarizes the technology gaps and recommended R&D for the GCR system concepts. The concepts included are those selected by the Generation IV International Forum in May 2002, as follows:

- International Near-Term Deployment concepts: Pebble Bed Reactor System (PBR) and Prismatic Modular Reactor System (PMR)

- Generation IV concepts: Very-high-temperature Reactor (VHTR) and Gas-cooled Fast Spectrum Reactor System (GFR).

The PBR and PMR also provide precursor R&D that supports the Generation IV reactor concept development. A great deal of the R&D (e.g., fuel, graphite, materials, components, nuclear physics, thermal hydraulics, disposal) is of generic importance for all GCR concepts and should be performed in a way that there is a cross-fertilization for mutual use and that the most stringent requirements are also covered. It is recommended that the generic part of the program be extracted in a next step and that a coherent approach be established on an international basis within the Generation IV International Forum.

The following is still organized by each of the four reactor system concepts and discusses technical or other types of known gaps, and the R&D or other activity recommended for accomplishment. Locations for accomplishing R&D activities where discussed are possibly, but not necessarily, unique facilities. The R&D matrix provides an overall summary of timing, costs, and priorities for the recommended R&D.

3.1 Pebble Bed Reactor System

The PBR has been under development for over twenty years in Germany. Much of the research has been done at the Juelich Research Institute. Advanced and larger versions of the PBR are being developed in South Africa and refinements are being studied at the Massachusetts Institute of Technology. Currently, a small pebble bed research reactor capable of producing electricity is undergoing startup testing in China (HTR-10). To gain the benefit of this technology for commercial application by being able to fully utilize the very high temperatures possible, gas turbines would have to be used with direct or indirect cycles. Indirect cycles would have to be employed if PBRs are to be used in process heat applications.

The technology gaps largely rest in the area of taking it beyond the small research stage to larger commercial applications that will require higher burnups, higher temperatures, and demonstrations of safety in terms of fuel performance, the availability of suitable materials, and dealing with accident scenarios. Detailed research plans for each will need to be developed with cooperating nations.

3.1.1 Fuel

While the performance of the uranium-oxide TRISO fuel was quite good in the German Arbeitsgemeinschaft Versuchsreaktor (AVR), operated from 1966 to 1988, the new demands for higher thermal efficiencies requiring higher core exit temperatures (>900°C), fuel temperatures up to 1300°C, and higher burnups from 80,000 to 120,000 MWd/MTU, require the development of new fuel particles. These higher operating conditions will also require new fuel safety tests to determine how such fuel responds to accident conditions.

The other major research area is associated with silver and cesium migration through the SiC. This becomes even more problematic at higher temperatures and burnups. The research is to identify the mechanism for diffusion and develop a means to limit such diffusion since it adversely affects normal operations and potential source terms in the event of accidental releases. This issue is more difficult for direct cycle machines. Should silver and cesium diffusion become a major problem, alternative coatings will need to be developed to limit the migration of these fission products.

As burnup increases and alternative fuels (e.g., plutonium) are considered, palladium attack of the SiC becomes important. Experiments to determine the extent of corrosive attack could limit the containment properties of the SiC. Research in this area is important for the higher burnups targeted. The R&D effort in the fuels area can be broken down into four major areas:

1. Develop and qualify existing and newly developed fuel

2. Develop and validate a fuel performance model to design advanced fuels

3. Identify and retard the mechanism of silver and cesium diffusion through the SiC and the impact of palladium corrosion or find other coatings

4. Test fuel under accident conditions.

Depending on the specific research area, this research could span 10 years at a total estimated cost of $42 million.

3.1.2 Fuel Cycle

The technology gaps in the fuel cycle area are focused on developing a fabrication process with an associated quality control and assurance program that can reliably produce high quality fuels that can withstand the high temperatures and fluence expected for the thermal reactors. In addition, R&D is required for used fuel volume reduction and waste disposal. Should a closed fuel cycle be economical, additional R&D will be required to reprocess and recycle microsphere fuel. The R&D effort can be broken down into the following major areas:

1. Develop a fabrication process for advanced microsphere fuel

2. Develop a quality control and assurance process for high volume manufacture

3. Develop a process for fuel volume reduction for disposal

4. Research the effects of direct disposal of fuel pebbles in repositories

5. Develop a fuel pretreatment process for recycling

6. Develop a process for manufacturing coated particle fuels and test the fuel similar to the Fuel tasks listed above.

Most of the research in the fuel cycle area can be completed with an aggressive program over a three to four year period at a cost of about $10 million. Should recycling be considered an option, this would extend the research program to 10 years at an additional cost of about $40 million.

3.1.3 Reactor Systems

The reactor systems research area is largely focused on the development of advanced materials that can withstand the high operating temperatures associated with these reactors. The research emphasis can be split into two parts: (1) find a means to insulate metals found in the piping and components, thus allowing operation within code allowable limits; and (2) develop improved materials that can accommodate higher temperatures.

The other major area in reactor systems is the performance of graphite in terms of lifetime and disposal. If high-temperature thermal gas reactors are deployed, a large volume of graphite will have to be disposed in both fuel and reflector form. Research areas are recommended for treatment of the graphite either for recycling or volume reduction. Associated with this effort is an understanding of the decommissioning issues for these reactors. Specific emphasis is needed on the following:

1. Control rod materials—new graphite fiber or composites
2. Reactor vessel higher temperature materials
3. Hot gas duct and other vessel materials
4. Recuperator designs and materials
5. Expansion joints to handle expected large thermal growths
6. Intermediate heat exchanger designs and materials to withstand accidents and transients
7. High temperature concrete for reactor cavity applications
8. Effects of helium and impurities on metal behavior at high temperature and irradiation
9. Qualifying graphite for full life
10. Research on how to volume reduce, recycle, or dispose of graphite
11. Developing an understanding of decommissioning issues based on the AVR and other decommissioned plants.

The reactor systems research could take from 2 to 10 years at a total cost of $50 million.

3.1.4 Balance of Plant and Energy Products

The most difficult challenge in the balance of plant area is the development of a helium-based power conversion system. Designs for direct-cycle machines rely on an immature technology that has not been demonstrated at the scale being proposed. Research on helium turbines and compressors is very limited as is work on magnetic and catcher bearings to handle such large loads. Indirect cycle machines have challenges associated with intermediate heat exchangers in materials, effectiveness, and transient behavior. To capture the high helium temperatures sought, turbine materials will need to be developed to avoid blade-cooling systems.

In the energy products area, hydrogen production facilities that are compatible with nuclear applications need to be designed. Specific research topics include:

1. Development of major components in the power conversion system
   - Catcher bearings
   - Compressors
   - Turbines
   - Magnetic bearings
   - Intermediate heat exchangers

2. Nuclear compatible process heat applications
   - Hydrogen
   - Desalinization
   - Interfacing systems to prevent cross contamination or safety consequences.

The balance of plant and energy products research would take from 2 to 10 or more years, especially if new high-temperature materials will be required for high-temperature turbines. The cost could reach $10.5 million.

3.1.5 Safety Concepts and Performance

The safety of PBRs is largely based on their low power density, high heat capacity, and natural ability to remove heat by conduction and radiative heat transfer without the need for active or passive safety systems. These conclusions presume good fuel performance, which is why the first task listed above is so important. LOCAs are generally assumed to be acceptable since core melt accidents can be deterministically excluded even without emergency core cooling. However, air ingress is a potential problem that needs better definition in terms of understanding the fundamental phenomenon and actual in-reactor performance without making overly simplified and conservative assumptions that result in requirements that negate some of the natural safety advantages of the technology. Once better understood, mitigation strategies need to be developed that still maintain the safety case not requiring additional safety systems.

Specific research topics include:

New Areas

1. Seismic compaction for reactivity insertions
2. Air ingress analysis for realistic plant conditions
3. Air ingress mitigation strategies
4. Thermal mixing
5. Fuel heat up tests at high temperatures and irradiation.

Confirmatory

1. Transient reactor models
2. LOCA analysis
3. Natural cooling capability of reactor
4. Pebble flow in core

5. Stochastic heat transfer in the pebble bed
6. Fission product transport
7. Probabilistic risk assessment model of plant.

Research in the safety area ranges from 2 to 10 years at a cost of $45 million.

3.1.6 Economics

Economics is perhaps the most speculative area of all Generation IV systems. Since the PBR is one of the designs that is closer to deployment, cost estimates should be closer to reality. Unfortunately, the South African design and cost information is largely unavailable and estimates have to be made. The research needs identified for the PBR are those associated with the advanced pebble bed being developed at the Massachusetts Institute of Technology, which has an indirect cycle system with components that are manufactured at factories in truck shippable modules that are site assembled. The strategy for maintenance is also based on replacement rather than repair due to the smaller equipment modules of the design. Also included in this concept are smart instrumentation and control systems that should reduce unplanned outages and staffing since the concept would rely on a control room design that utilizes expert systems for operations and control.

Specific topics include:

1. Equipment monitoring system development
2. Advanced control room designs for multi-plant operation with expert systems
3. Modularity concept development to allow for factory fabrication and minimum site work
4. Smart system development to enhance component reliability
5. Site staffing optimization by designing plant to minimize use of on site staff with centralized maintenance organizations.

Most of these research areas are medium-term (2 to 5 years) with costs up to $4.5 million.

3.1.7 Security

The PBR has the advantage of on-line refueling that allows for higher capacity factors. While an advantage in power generating capability, it presents a potential for on-line diversion of pebbles. Although studies have been done that show that hundreds of thousands of pebbles would be required to be diverted to accumulate any meaningful quantities of plutonium, additional safeguard systems would be required for fuel accountability. The results of this research would be included in the design of the facility.

Additionally, since the terrorist attack on the World Trade Center, additional research is needed to determine what design changes might be required to minimize the potential impact of possible terrorist scenarios. Both of these are short-term to medium-term analysis efforts requiring about $2 million.

3.1.8 Major Codes

Since the PBR was developed in Germany, much of the computer code work originated there. The major neutronics and thermal hydraulics code of record is Very Special Old Programs (VSOP). VSOP is capable of handling the on-line refueling of the pebble bed core. In addition, the space time kinetics code (TINTE) is also not of U.S. origin or familiarity. Accident and transient analysis codes are typically modifications of LWR codes, such as RELAP and RETRAN, that may not be suitable for gas reactor application and are not benchmarked even if suitable. In short, for the U.S. considerable computer code development and benchmarking will be required for confidence in predicting the safety performance of the plant.

Specific research areas include:

1. Qualify VSOP (verification and validation effort)
2. Develop a replacement for VSOP
3. Develop and benchmark a LOCA code
4. Develop and benchmark an air and water ingress code
5. Develop and benchmark a reactor kinetics code (TINTE)
6. Develop an integrated plant model of the primary and secondary side
7. Develop a reactivity insertion code package
8. Develop a fission product retention and release code.

These R&D efforts are mid- to long-term (10 years), depending on the code package. Costs to develop these codes could exceed $23 million.

3.1.9 Summary

The total R&D program for an advanced high-temperature PBR that can achieve the high-temperature and high-burnup targets with the demonstrated safety attributes that would allow the plant to be built economically is close to $200 million. The South African project is aimed at less lofty goals, which is why their demonstration program is considerably less costly. Much can be learned from their project should it move forward.

3.2 Prismatic Modular Reactors

This section describes the development plan to support the design of the PMR, including the GT-MHR and PMR concepts for process heat applications, such as the hydrogen-producing modular helium reactor (H2-MHR).

Although a considerable gas-cooled technology base is available, as demonstrated by more than 50 GCRs built and operated around the world and in the United States since 1956 (including the Peach Bottom and Fort St. Vrain reactors as well as the Japanese HTTR), stricter regulatory requirements and more advanced requirements for the operating environment necessitate further development programs and, to some degree, the reestablishment of past technology development. These programs are foremost in the areas of fuel development, metallic materials, and graphite and ceramic materials, including technology development, which provide data for design methods and validation of computer codes. Engineering development is also needed for component or process verification, including prototypical component testing.

Each section in the development plan summarizes the technical activities within a specific development area as well as cost and schedule information.

3.2.1 Fuel Development

The fuel development activities outlined in this section represent a plan to reestablish modular helium reactor (MHR) fuel manufacturing capability in the U.S. based on German coating technology, and to expand the existing fuel performance and fission product transport data base to support design and licensing.

The fuel for the MHR consists of TRISO-coated uranium oxycarbide (UCO) fissile particles and TRISO-coated UO_2 (or UCO) fertile particles. The TRISO coating surrounding the fuel kernel consists of a low-density buffer coating, which in turn is enclosed in a SiC coating contained between an inner and outer pyrolytic carbon (IPyC and OPyC) coating. The TRISO coating forms a strong pressure vessel for containing both gaseous and metallic fission products and is the primary barrier to the release of radionuclides. The fuel particles are bonded together with a carbonaceous matrix to form cylindrical graphite compacts, which are assembled into blind fuel holes in hexagonal graphite blocks that are about 14.8 in. (376 mm) across flats and about 31 in. (787 mm) in height.

The MHR fuel draws upon fuel fabrication experience demonstrated in the U.S., the U.K., Germany, and Japan over the past 35 years. In Germany, substantial quantities of coated particle fuels have been fabricated by NUKEM since 1965. More than 10,000 kg of coated particles were made for the AVR and Thorium High-Temperature Reactor (THTR) before the NUKEM fuel manufacturing facility was decommissioned in 1990. In the U.S., fuel assemblies for the Peach Bottom Unit 1 and the Fort St. Vrain initial cores and reloads were manufactured at the General Atomics (GA) fuel production facility in San Diego. For two Peach Bottom cores, about 3,500 kg of pyrocarbon-coated fuel particles were fabricated into more than 1,600 fuel element assemblies. For the Fort St. Vrain initial core and three reload segments, about 30,000 kg of TRISO-coated uranium and thorium carbide particles were fabricated into 2,250 fuel element assemblies.

The manufacturing processes for the MHR, however, must be capable of producing fuel of higher quality than Fort St. Vrain to satisfy more restrictive quality and irradiation performance requirements, which means reducing the defective particle fraction in as-manufactured fuel. The results of the extensive fuel qualification program carried out in the German GCR program have demonstrated the feasibility of fabricating high-quality fuel capable of meeting MHR performance requirements. Coated fuel particles tested in support of the Japanese HTTR program have also shown good performance for fuel with low defect levels at low burnups and moderate fast fluence.

Much of the German irradiation data is for UO_2 fuel irradiated to burnups that are substantially lower than the design burnup for MHR fissile particles; however, there is compelling theoretical and experimental evidence to support the superiority of UCO fuel for reactor service to higher burnup. This is also substantiated by the excellent performance of UCO fuel in German capsule tests (FRJ2-P24) using German coating technology. Nevertheless, because irradiation performance data for TRISO-coated UCO in fuel compacts is limited and because there is essentially no post-irradiation heating test data for high-quality, high-performance UCO fuel, irradiation testing and post-irradiation accident simulation tests are included in the MHR fuel plan to demonstrate and qualify UCO fuel for MHR service conditions and accidents.

The following objectives must be satisfied to meet the overall objective of qualifying the fuel and the fuel fabrication processes for the GT-MHR prototype plant:

- Fabricate, irradiate, and perform post-irradiation simulated accident conditions testing of *demonstration test* fuel to demonstrate that TRISO-coated UCO fuel manufactured under reference process conditions and meeting Fuel Product Specification requirements performs satisfactorily under MHR normal operating conditions and accident conditions

- Fabricate, irradiate, and perform post-irradiation simulated accident conditions testing of *qualification test* fuel to verify that TRISO-coated UCO fuel manufactured under reference process conditions and meeting Fuel Product Specification requirements performs satisfactorily under the full range of MHR normal operating conditions and accident conditions

- Conduct fuel testing and fission product transport technology development, to the extent necessary, to generate the data needed to validate fuel performance and fission product transport models in support of licensing of an MHR prototype plant

- Design and construct a fuel fabrication pilot plant consisting of production scale process equipment

- Demonstrate that fuel meeting MHR fuel product specification requirements can be manufactured in the pilot plant

- Fabricate, irradiate, and perform post-irradiation accident conditions testing of *proof test* fuel to verify that the pilot plant fuel performs satisfactorily under GT-MHR normal operating conditions and accident conditions.

The fuel plan work scope is divided into five major work areas: (1) demonstration test fuel fabrication and irradiation, (2) fuel fabrication process improvement, (3) fuel manufacturing pilot plant, (4) fuel qualification, and (5) fission product transport technology development.

3.2.1.1 Demonstration Test Fuel Fabrication and Irradiation. The fuel qualification strategy based on German coating technology is considered to offer the highest probability for success. It is necessary to demonstrate that the German coating process, when transferred to the U.S., will yield coated UCO and UO_2 particles that perform well when formed into compacts and subjected to MHR normal and potential accident conditions.

A fuel demonstration test (known as MHR-1) is scheduled to be irradiated in the High Flux Reactor (HFR) at Petten in the Netherlands starting in late 2002. This capsule will contain fuel compacts fabricated from German TRISO-coated UO_2 proof test particles using GA's reference thermoplastic (TP) matrix-based compacting process. This test will demonstrate that the GA compacting process is not detrimental to the demonstrated excellent irradiation performance of the German fuel particles. The test is also intended to show that the high-quality German fuel particles will perform well in high packing fraction compacts as well as in the more lightly loaded German spherical fuel elements. The irradiation conditions for MHR-1 (e.g., temperature, fast neutron fluence, and burnup) will be consistent with the irradiation conditions to which the German fuel particles were subjected in German irradiation tests. The MHR-1 will provide a direct comparison of the performance of the high-quality fuel particles in GA-fabricated compacts and in German-fabricated pebbles.

The scheduled MHR-1 irradiation test in Petten will take approximately 3½ years to complete. The cost of participation and data analysis is estimated at $1.0 million.

3.2.1.2 Fuel Fabrication Process Improvement. With the decommissioning of both the fuel manufacturing facility and the developmental fuel fabrication facility at Fort St. Vrain, GA no longer has facilities in which to fabricate MHR fuel. It is, therefore, necessary to reestablish fuel fabrication capability by setting up equipment to fabricate qualification test fuel prior to construction and operation of a fuel manufacturing pilot plant. This will be accomplished by assembling a pilot line consisting of relatively small-scale UCO and UO_2 kernel lines (for fissile and fertile kernels, respectively), a full-size production coater, and laboratory scale compact fabrication equipment. The pilot line will be set up in a facility that is of sufficient size to be expanded into the pilot plant for fabrication of the fuel for the prototype MHR. It is expected that the facility will be located on a government site, such as Savannah River, Hanford, or Oak Ridge National Laboratory (ORNL), that has the necessary infrastructure, including essential resources such as Health Physics, a nuclear material control and accountability system, and so forth.

3.2.1.2.1 Kernel Process Development—The reference kernel process for fabrication of UCO kernels for the fissile MHR fuel particles is the internal gelation process initially developed at ORNL. Babcock & Wilcox (B&W), now BWXT, used this process to make large quantities of 200-μm kernels for the New Production Reactor program in the early 1990s. B&W also fabricated a small number of 350-μm kernels for the DOE commercial GT-MHR project in 1994. MHR fertile particles can be either UCO or UO_2, but it is anticipated that UO_2 kernels will be used because they are more easily manufactured than UCO kernels. The main efforts associated with the UCO kernel process are to optimize the process conditions for making 350-μm kernels and to qualify a substitute for trichloroethylene (environmentally unacceptable) in the process. In addition, data needed are batch sizes, times, yields of acceptable product, and quantities of fuel material recovered from scrap for further processing.

3.2.1.2.2 Fuel Coating Process Development—The objective of the coating process work is to successfully replicate the German coating technology within a process suitable for large-scale fuel production. This will be accomplished by designing and installing a coater that provides a coating environment equivalent to the coating environment in the German production coater, and that has appropriate features for loading, unloading, cleaning, etc. Test runs would be made with this coater to establish the process parameters for coating 350-μm kernels (extrapolating from the process parameters used by the Germans to coat 500-μm kernel) to determine the maximum batch size consistent with product quality requirements, and to evaluate the economics and product quality associated with straight-through coating.

3.2.1.2.3 Fuel Compact Fabrication Process Development—Currently, GA's reference fuel compact fabrication process is the thermoplastic (TP) matrix-based process used in Fort St. Vrain fuel manufacturing. This process was optimized for very high quality MHR fuel production in 1994 and 1995 under the DOE-Office of Nuclear Energy commercial GT-MHR Program, and is to be qualified for this purpose in irradiation test MHR-1. However, for large-scale fuel manufacturing, a thermosetting (TS) matrix-based process is preferred for several reasons. First, the TS matrix-based process would result in improved fuel quality because the TS matrix would be formulated from raw materials having lower levels of impurities than the TP matrix; the TS matrix would yield stronger, less friable compacts; and the TS matrix process would involve less handling of the compacts, thereby reducing the potential for damage. Second, the TS matrix-based process would involve fewer steps and be better suited to automation, which would reduce the cost of fuel compact fabrication. Consequently, GA plans to substitute a TS matrix-based process for the TP matrix-based process as the reference compact fabrication process, provided that compacts fabricated using a TS matrix-based process perform well. Operational test data are needed on batch sizes, compacting times, yield of acceptable product, and quantities of fuel material recovered from scrap for further processing.

3.2.1.2.4 Quality Control Test Technique Development—Improved quality control (QC) techniques are needed to demonstrate that the MHR fuel will meet the stringent quality requirements with high confidence. These consist of improved techniques for characterization of fuel, including the detection of SiC defects, characterizing the microstructures of IPyC and OPyC layers, direct measurement of permeability of the IPyC coating layer, and stoichiometry of UCO kernels. The QC techniques should be automated to improve the reproducibility and decrease the time required for measurements consistent with large-scale production plant requirements. To the largest extent possible, these methods should also be nondestructive, capable of high throughput rates (potentially high enough to make 100% inspection feasible), and capable of providing near real-time feedback to the fuel fabrication processes.

The improvement in QC methods is also necessary for reducing the cost of QC and the reduction of radioactive waste.

3.2.1.2.5 Fuel Product Recovery Development—A fully qualified fuel manufacturing process requires the development of techniques to recover or dispose of uranium material generated during each manufacturing step, provide adequate accountability for uranium material, and convert radioactive waste streams for re-use or disposal. This requires the development of improved software and procedures to provide real-time data on the quantities of uranium in all process streams.

3.2.1.2.6 Cost and Schedule—The fuel fabrication process development activities support the design and construction of the fuel fabrication pilot plant and must be completed prior to start of fuel fabrication for fuel irradiation proof testing. The cost estimate includes costs of all process equipment design and construction, and cost of fuel production for irradiation testing.

Table 4. Fuel fabrication process improvement R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Kernel Process Development											1,700
Fuel Coating Process Development											6,100
Fuel Compact Fabrication Process Development											3,500
Quality Control Test Technique Development											7,500
Fuel Product Recovery Development											1,500
Total											20,300

3.2.1.3 Manufacturing Pilot Plant. The pilot plant for fabrication of the initial core for the GT-MHR prototype plant will be sized to produce approximately 450 fuel elements per year based on a reload frequency of 16 months for the two-segment MHR core. As discussed above, it is expected that the pilot plant will be located on a government site, such as Savannah River, Hanford, or ORNL, and that the facility will be designed first to accommodate the pilot line, then be expanded into the pilot plant. Expansion of the pilot line into the pilot plant will start immediately upon successful completion of the MHR-1 irradiation (based on fission gas release results).

The first subtask under this task will be to perform a cost evaluation to develop estimates of the fuel unit cost for a commercially viable MHR fuel manufacturing plant and the fuel unit cost potentially achievable in a fuel manufacturing plant utilizing the reference manufacturing processes. These cost estimates are needed to provide guidance to the pilot plant design effort and to identify process modifications that may need to be made to reduce manufacturing costs to acceptable levels. An objective of the design effort will be to achieve a unit fuel production cost in the pilot plant that is low enough to demonstrate the feasibility of reaching a competitive fuel cost in a commercial fuel manufacturing plant.

After completion of pilot plant construction, the pilot plant will be operated for approximately nine months to shake down the equipment, train the operating staff, and demonstrate that the fuel fabrication processes are repeatable and capable of manufacturing fuel that satisfies fuel product specifications. Following this demonstration of the fuel fabrication processes, the proof test fuel will be fabricated.

Following proof test fuel fabrication, the manufacturing processes will be "frozen." Subsequent changes to the reference processes will require fabrication and irradiation testing to demonstrate that the process changes do not negatively impact fuel performance. It is envisioned that a continuing fuel fabrication process improvement program will be conducted in the pilot plant following fabrication of the fuel for the prototype plant initial core, and that the improved processes resulting from this program will be qualified via fuel irradiation testing in the prototype MHR. This continuing fuel process improvement program will eventually result in an economically viable fuel manufacturing capability for the n^{th} MHR plant.

The cost estimate to design, fabricate, and operate the pilot plant has not been developed.

3.2.1.4 Fuel Qualification. The work scope for fuel qualification includes a series of irradiation tests and post-irradiation heating tests to qualify the fuel product and fuel manufacturing processes for MHR normal operation and credible accident conditions, and to obtain the data needed to improve and validate the existing fuel performance models.

This program will consist of the planning, design, execution, and analysis of capsule irradiation experiments, post-irradiation examinations (PIE), and out-of-reactor isothermal heating tests. The planned tests will ensure that statistically significant numbers of particles are exposed to conditions that simulate both normal reactor operation and conditions that include normal operation plus an operating margin. Following irradiation, the test capsules will be moved to remote handling facilities where the capsules will be disassembled to obtain samples for PIE and post-irradiation heating tests. The fuel samples will undergo PIE to determine fuel performance as a function of operational parameters, including particle failure, fission product retention, and fuel failure mechanisms.

3.2.1.4.1 Qualification Test Irradiation and PIE—A series of three irradiation test capsules are tentatively planned to qualify the fuel product and fuel manufacturing processes after the acceptable performance of the fuel has been demonstrated in irradiation test MHR-1. These tests include: (1) a test at average MHR core conditions, (2) a test under bounding MHR conditions, and (3) a margin test (to temperatures of about 1400°C and a fast neutron fluence around $5 \times 1025\ n/m^2$) to demonstrate an adequate safety factor. These qualification tests will provide the statistical database necessary to establish and validate the fuel performance models for coated particles with low failure fractions over the range of MHR normal operating conditions and simulated accidents. If one or more of the qualification tests are individually purged multi-cell capsules, they can potentially include test samples of as-manufactured defective particles and other materials needed to obtain data for fuel design, validate the models for fuel performance of the small fraction of defective fuel particles, and validate fission product transport computer codes. Otherwise, one or more additional capsules may be required for irradiation of these samples.

3.2.1.4.2 Proof Test Irradiation and PIE—The final test is a proof test of fuel fabricated in the pilot plant using full-scale production equipment. A fuel proof test is needed to assure that the fuel product specification, fuel manufacturing process, and fuel design have been adequately defined, and reliably produce a fuel that performs in accordance with design requirements and fuel performance models. Data (fuel failure fraction as inferred from Kr-85m and Xe-133 release and metallic release [Cs-137]) are needed to confirm that fuel from the full-size fabrication equipment meets performance requirements. Sufficient post-irradiation heating data needs to be obtained to confirm that the performance of the irradiated proof test fuel under accident conditions is predicted by results obtained on qualification test fuel.

3.2.1.4.3 Post Irradiation Heating Tests—Particles irradiated in the demonstration, qualification, and proof test capsules will be subjected to out-of-reactor isothermal heating tests to simulate conduction cool-down accident conditions. The tests will cover a range of parameters bounding postulated accident conditions and will ensure that a statistically significant database is developed. The fuel specimens will be characterized following the heating tests to determine fuel performance and fission product retention. The database from the accident testing will be combined with the existing international database on fuel and fission product behavior under accident conditions and used to upgrade and validate fuel performance models for accident conditions.

3.2.1.4.4 Fuel Performance Model Development and Validation—The irradiation tests provide the database necessary to establish and statistically validate the fuel performance models and codes for predicting fuel performance under the full range of MHR normal operating conditions and simulated accidents. The data needed include:

- *Coating Material Property Data*—Fuel performance depends on the mechanical and thermal properties of pyrocarbon and SiC, such as bulk density, porosity distribution, and crystallite anisotropy. For pyrocarbons, the failure fraction of IPyC and OPyC coatings are needed as a function of fluence at temperature. For SiC, coating strength as a function of fluence at temperature and fission product transport behavior, are needed. Data are needed over a range of exposure conditions that encompass the reactor operating envelope. These data provide the basis to select the pyrocarbon and SiC characteristics for the fuel specification.

- *Defective Particle Performance Data*—Failure of defective particles (particles with as-manufactured defects) is predicted to be a major contributor to fission product release from the MHR core during normal operation and postulated accidents. Single-effects data on the performance of defective particles are needed to refine fuel particle performance models including the retention of fission products. The coating failure fractions provide some information to estimate or bound the performance, especially if the performance of failed defective particles due to missing buffer coatings, missing or defective SiC coatings with intact OPyC, missing or failed OPyC coatings, and heavy metal dispersion in the buffer coating (IPyC defects) can be characterized. Data from a fuel experiencing different ratios of thermal to fast flux are needed to separate the effects of burnup and fast fluence.

- *Thermochemical Performance Data for Fuel*—The thermochemical phenomena that establish the ultimate thermal performance limits of TRISO particles have been identified as: (1) kernel migration, (2) fission product/SiC coating interactions, and (3) thermal decomposition of the SiC coating. All of these phenomena are expected to occur in TRISO-coated UCO fuel at sufficiently high temperatures, high thermal gradients, and long times. Kernel migration rates are required for TRISO-coated UCO particles as a function of temperature, thermal gradient, time, and kernel composition. The rates of fission product/SiC interactions need to be determined as a function of temperature, thermal gradient, and time. It is particularly important to separate the temperature and thermal gradient dependencies and to determine the time dependence because these variables are of particular importance in extrapolating the results of relatively short, accelerated irradiation tests to predict in-core performance, which is characterized by lower thermal gradients but longer times. The rates of thermal degradation of irradiated TRISO particles under core heatup conditions need to be determined as a function of temperature, thermal gradient, and time. The appropriateness of using cesium release as the exclusive indicator of SiC failure needs to be confirmed.

- *Fuel Compact Thermophysical Properties*—The thermophysical properties of unirradiated and irradiated reference fuel compacts need to be characterized to support the validation of the core design methods for determining the distribution of thermal stresses and temperatures in the fuel compacts. Heat capacity, thermal expansion, and thermal conductivity are required as a function of shim content, matrix density, fast neutron fluence, and temperature to generate the fuel performance models.

3.2.1.4.5 Cost and Schedule—The costs of fuel irradiation, post irradiation examination, and out-of-reactor isothermal heating test activities do not include the fabrication and preparation of fuel test samples. These are included in the fuel fabrication process development costs.

Table 5. Fuel qualification R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Qualification Test Irradiation and PIE	■	■	■	■	■						15,000
Proof Test Irradiation and PIE			■	■	■						10,000
Post Irradiation Heating Test		■	■								8,000
Fuel Performance Model Development Data		■	■	■	■						8,500
Total											41,500

3.2.1.5 Radionuclide Transport. Fission product transport development activities aim to reduce the uncertainty in existing computer models and databases used in reactor design to predict the release and transport of gaseous and metallic fission products. This is achieved by obtaining data from (a) fission product irradiation capsule tests and PIE, (b) post-irradiation heatup tests of fuel particle samples from capsule tests, (c) single effects testing under representative MHR conditions, and (d) integral testing. The data from these tests provide input to radionuclide transport models and methods validation. Included in these tasks is also the data acquisition for the development and validation of corrosion models for TRISO-coated particles, fuel compacts, and boronated control materials.

Although a significant amount of data exists from prior GCR fuel irradiation programs, additional fission product data for MHR fuel are required for further fission product model refinement and methods validation, as follows.

3.2.1.5.1 Radionuclide Transport in the Reactor Core—

- *Fission Gas Release from Fuel Particles*—The dominant sources of fission gas release, including iodine and tellurium isotopes, are uranium contamination in the fuel compact matrix and failed fuel particles with exposed UCO kernels. The release characteristics of these two sources need to be determined. Data are needed on fission gas release rates (Kr, Xe, I, and Te) as a function of temperature, half-life, burnup, and flux under irradiation and under dry and wet core conduction cooldown conditions. In addition, the effect of hydrolysis on gas release must be quantified for steady-state irradiation and for transient wet core conduction cooldown conditions. Single effects tests will be performed on laser-failed fuel particles.

- *Fission Product Diffusivities in Fuel Kernels and Particle Coatings*—The fuel kernel of the coated particle and the fuel particle coatings, particularly the SiC coating, are the initial barriers to the release of fission metals from the core and may provide significant holdup, especially in low-burnup kernels. Correlations are needed for the effective diffusivities of key fission metals (Cs, Ag and Sr) and Pu isotopes in LEU/natural UCO fuel kernels and coatings as a function of temperature, burnup and neutron flux for normal operation and dry and wet core conduction cooldown conditions. The data will be obtained from PIE-failed fuel particles from capsule tests and post-irradiation heating tests.

- *Fission Product Diffusivities/Sorptivities in Graphite* —The fuel element graphite can significantly attenuate the release of fission metals and preclude the release of actinides from the core during normal operation and during core conduction cooldown transients. Correlations/models for the diffusivities and sorptivities of Cs, Sr, Ag, and Pu isotopes in fuel-compact matrix and core graphites as a function of temperature, fast fluence, coolant impurities, system pressure (for Ag), and the extent of graphite oxidation under normal operating and dry and wet accident conditions are needed. Out-of-pile and in-pile single-effects tests of irradiated and unirradiated matrix test specimens will be conducted.

3.2.1.5.2 Transport in the Primary Coolant Circuit—

- *Radionuclide Deposition Characteristics on Structural Metals*—Condensable radionuclides, including iodines and volatile fission metals, released from the core during normal operation and during certain accidents are transported in the primary coolant circuit and will tend to deposit in the power conversion unit (PCU), thereby attenuating their release to the environment. However, this plateout activity is a major contributor to the occupational exposure during maintenance and in-service inspection. Data are needed to characterize the deposition of Cs, Ag, I, and Te on metals. Correlations that give the sorptivities of these nuclides as a function of temperature, partial pressure, surface state, and coolant chemistry for normal operating and accident conditions are needed. These sorption data should be obtained at representative partial pressures to avoid the orders-of-magnitude extrapolations necessary with the present database. Particular attention should be given to the effects of H_2O and dust on the deposition process and to the possibility of chemical reactions involving radionuclides under accident conditions (e.g., CsI formation). The diffusivities of silver and cesium in the PCU structural metals are needed under normal operating conditions, with special attention to the effects of surface films, to determine whether or not indiffusion must be explicitly modeled. These data will be obtained from single-effects tests in a high-pressure, high-flow, out-of-pile loop under representative service conditions.

- *Characterization of the Effect of Dust on Radionuclide Transport*—The presence of circulating and/or deposited particulate matter (dust) in the PCU may alter the plateout distributions during normal operation and may increase the extent to which condensable radionuclides are released from the PCU during dry and wet depressurization transients. Data from laboratory studies that elucidate the effects of dust on the transport of condensable radionuclides in the PCU during normal operation and during transients, especially the effects on the re-entrainment/redeposition characteristics during dry and wet depressurization transients are needed. A prerequisite to these measurements is the determination of the representative dust (chemical composition, concentration, particle size distribution) in a prismatic core.

- *Radionuclide Re-entrainment Characteristics for "Dry" Depressurization*—Radionuclides, which deposit in the PCU during normal operation, may be partially re-entrained and released from the primary circuit during primary coolant leaks. The extent to which plated out activity may be removed during rapid depressurization transients needs to be quantified. Correlations that give the fractional liftoff of I, Sr, Cs, Te and Ag as a function of the controlling system parameters are required. Test variables that must be investigated include shear ratio, absolute wall shear stress, blowdown duration, temperature, humidity, and surface oxidation state.

3.2.1.5.3 Radionuclide Transport in the Containment Structure—

- *Fission Product Transport in a Vented Low-Pressure Containment*—The vented, low-pressure containment (VLPC) could be a significant barrier to the release of radionuclides to the environment during depressurized core conduction cooldown transients. During primary coolant leakage into the reactor building, natural processes will act to reduce the level of entrained radionuclides as the gas stream transits the building. Data are needed for the condensation, settling, and plateout of I, Cs, Te, and Ag on reactor building construction materials. The effects of temperature, coolant chemistry, surface state, and aerosols must be treated explicitly. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) also needs to be determined with particular attention to the effects of coolant chemistry on composition. The extent to which LWR data on radionuclide transport, especially transport in a containment building, are applicable to the MHR VLPC needs to be determined. Laboratory-scale single effects scoping tests will be conducted.

- *Decontamination Efficiency of Pressure Relief Train Filter*—During a hypothetical event, which combines large water ingress with loss of forced cooling and failure to terminate water ingress, the primary helium relief valves could become a release pathway for primary coolant and any entrained radionuclides. To limit the consequences of such an event, the VLPC design includes a piping network and a filter, which will act to decontaminate the gases released through the valve(s). Data are needed to validate the design methods describing the filter DF and possible re-entrainment of radionuclides deposited on the filter under wet (following water ingress events) and dry depressurized core conduction cooldown conditions. The effects of temperature and coolant chemistry state must be treated explicitly. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) must also be determined, with particular attention to the effects of coolant chemistry on composition.

3.2.1.5.4 Core Corrosion—

- *Coated B_4C Corrosion Data*—The pyrocarbon-coated B_4C granules in the reserve shutdown control pellets may be corroded by coolant impurities, principally H_2O and oxygen, which could compromise the reactivity control capability. The exposed B_4C may hydrolyze and the resulting boric acid, which is quite volatile, may be lost from the core. Data are needed describing the corrosion of PyC-coated B_4C granules by coolant impurities during normal and off-normal operating conditions including water ingress events. The corrosion rate must be determined as a function of time over a range of fixed temperatures, impurity concentrations, and system pressure. The temperature below which the oxidation reaction is not mass-transfer limited must be confirmed. The boron vapor species should be characterized.

- *Core Matrix Materials Corrosion Data*—The carbonaceous matrix materials used as binders in the fuel compact, lumped burnable poison, and the reserve shutdown control pellets consist of finely divided graphite flakes bonded together with residual carbon from the carbonized pitch binder. The matrix may be corroded by coolant impurities, principally H_2O and oxygen. Extensive corrosion of these materials could potentially lead to loss of structural integrity for the fuel compact, with reduced thermal conductivity and associated high temperatures. The transport of coolant impurities and corrosion products in these matrix materials must be quantified for normal operating conditions and off-normal conditions, including water ingress events.

3.2.1.5.5 Cost and Schedule—The cost of the test programs for obtaining radionuclide transport and corrosion data are shown in Table 6. These estimates assume the use of existing hot cell facilities and in-pile and out-of-pile gas loop facilities (except modifications to perform tests). All fission product transport testing and code validation activities need to be completed one year prior to fuel load of the first Module.

Table 6. Radionuclide transport R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
FP Single Effects Tests	■	■	■	■							8,200
FP Integral Tests			■	■	■						5,300
Post Irradiation Heat-up Tests				■	■						3,600
Corrosion Tests				■	■						800
Total											17,900

3.2.2 Thermal Hydraulics Development

Satisfactory GT-MHR thermal hydraulic performance during normal operation requires accurate knowledge of the flow distribution and pressure drops through the core, lower plenum, and hot duct. The design is based on calculated fuel and control rod temperatures, resultant fuel and reflector block stresses, and core exit coolant temperature distributions (hot streaks) that are impressed on the hot duct and gas turbine. Fluid flow data are needed as input to the computer codes that perform these calculations.

- *Fuel Element Channel Flow Data*—Most of the core flow passes through the fuel element coolant channels. The coolant channel friction factor is needed to ensure the core pressure drop allocation is met, and that the desired partition of flow between the coolant channels and control rod channels is achieved. The friction factor data need to be obtained for representative drilled graphite coolant channels under normal reactor operating conditions.

- *Control Rod Channel Flow Data*—The design of the control rod channel is such that most of the pressure drop takes place through small flow passages at the channel entrance and exit to prevent large bypass flow through these channels when the control rods are withdrawn. Data are needed to determine the channel entrance and exit flow loss coefficients and the pressure loss in the channel when the control rod is inserted to ensure adequate cooling flow to the control rods.

- *Core Pressure Drop and Flow Mixing Data*—The computer codes that calculate flow maldistribution in the reactor core need validation of the following parameters: core pressure drop, maldistribution of coolant channel flow in the columns, the temperature of the coolant entering the hot duct, and the temperature of hot and cold streaks entering the gas turbine. Data are needed to obtain the core metallic plenum element and top reflector pressure drop and flow distribution, and bottom reflector/core support pressure drop and flow mixing.

- *Core Crossflow Test Data*—Transverse flow between core columns can cause flow maldistribution in the fuel blocks resulting in increased coolant temperatures, increased fuel temperatures and/or higher thermal stresses in the graphite blocks. Data are needed for loss coefficients for standard, reserve shutdown control, and reflector control rod block crossflow gaps as a function of expected crossflow pressure differentials, crossflow gaps, and coolant and bypass gas Reynolds numbers.

The specified data needs will be obtained from airflow tests with full-scale mockups of partial reactor core configurations. This will require an existing airflow test facility. The testing to obtain the core thermal hydraulic data can be completed inside a two-year period. The total cost of these tests is estimated at $2.9 million assuming no facility costs.

3.2.3 Metallic Materials

The structural materials development programs provide test data on material behavior and strength, including environmental effects on materials selected for use in MHR components. Three major components have structural materials data needs: reactor internals (core support upper plenum shroud, upper core restraint, and hot duct made from Alloy 800H), reactor internals structural ceramics (aluminum oxide used as thermal insulation), and reactor vessel (pressure vessel plate and forging made from modified 9Cr-1Mo-V ferritic steel, and bolting material made from Alloy 718).

3.2.3.1 Reactor Metals. Structural integrity of the metallic reactor internals and hot duct components is required to support the reactor core and protect the primary pressure boundary from overheating during conduction cool-down events. Furthermore, removal and replacement of these components prior to the completion of the reactor design life would severely affect plant availability.

The metallic components are exposed to neutron irradiation and to the temperatures and chemistry of the primary coolant during the life of the plant. Although significant data exist to quantify these effects on the mechanical properties of Alloy 800H base metal and weldments, testing will continue with emphasis on long-term neutron exposure and thermal aging.

- *Irradiation Effects on Metallic Reactor Internals Materials*—Data are needed on the effects of neutron irradiation during operating and conduction cool-down accident conditions on the properties of Alloy 800H base metal and weldments, including tensile strength, low cycle fatigue strength, fracture toughness, creep and relaxation data, creep-fatigue strength, and high-cycle fatigue strength.

- *Effects of Primary Coolant Chemistry and Temperature on Hot Duct Materials*—Data are needed on the effects of elevated temperature corrosion from primary coolant helium impurities during operating and conduction cool-down accident conditions on the properties of hot duct Alloy 800H base metal and weldments, including tensile strength, low-cycle fatigue strength, fracture toughness, creep and relaxation data, creep-fatigue strength, and high-cycle fatigue strength.

Tests will be performed on specimens to provide sufficient data to quantify the effects of neutron irradiation and primary coolant chemistry on time-dependant and time-independent mechanical properties of Alloy 800H base metal and weldments. An irradiation facility is required to irradiate the metallic specimens. Hot cells with tensile and creep test machines are required for the material property testing.

It is anticipated that the testing can be accomplished within 2½ years. The cost estimates assume no cost for facilities and testing machines.

Table 7. Reactor metals R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Irradiation Effects on Metallic Reactor Internals Materials											3,730
Effects of Primary Coolant Chemical and Temperature on Hot Duct Materials											3,070
Emissivity on Metallic Reactor Internals Materials											1,500
Total											8,300

3.2.3.2 Vessel Materials. The reactor vessel will be fabricated from modified 9Cr-1Mo-V ferritic steel per SA-387 Grade 91, Class 2 (for plates) and SA-336 Grade F91 (for forgings). Bolting material will be high-temperature Alloy 718, austenitic nickel-iron-chromium-molybdenum-niobium alloy SB-637. The database characterizing the effects of fast neutron ($E > 0.1$ MeV) irradiation on the nil ductivity transition temperature (NDTT) and other mechanical properties of 9Cr-1Mo-V needs to be extended since the material is not yet approved for nuclear vessel applications above 700°F under the ASME, Section III. Only limited irradiation effects data are available for Alloy 718 bolting material.

- *Irradiation Data for Reactor Vessel Materials*—The database for determining the neutron-induced property changes for Modified 9Cr-1Mo reactor vessel materials must be expanded to cover GT-MHR conditions. This includes data to characterize the neutron-induced changes in fracture toughness, tensile, and creep properties for the reactor vessel plate and forging materials, weldments, and heat-affected zone at reactor irradiation temperatures and neutron flux, fluence, and expected spectrum levels. Key materials variables include chemical composition (e.g., Cu, Ni, S, P, V, N), product form and size, and heat treatment. Mechanical property and creep data are required to establish whether irradiation has a significant impact on time-dependent properties as well as on NDTT shift. The locations of concern for which data are needed include the reactor vessel beltline, main flange, and closure head.

- *Properties of Heavy Section Vessel Materials Forging and their Weldments at Elevated Temperatures*—An insufficient property database exists on forgings and their weldments for the reactor vessel fabricated from Modified 9Cr-1Mo, SA-336 Grade F91 material. The essential data on material properties to support component design up to 460,000 hours service life are needed, including creep-rupture, isochronous stress-strain curves, creep-fatigue interactions, aging factors, and weld strength reduction factors. The database must be assembled and analyzed to establish allowable stress intensities and other required properties for submittal to the ASME.

- *Reactor Vessel Emissivity*—Reactor vessel emissivity is an important parameter in the removal of residual and decay heat during core conduction cooldown events. To satisfy radiological release criteria and their associated fuel temperature limit, a minimum emissivity of 0.8 is required of the reactor vessel materials throughout its operating life. If the emissivities do not satisfy the design requirement, methods of surface treatments, such as surface roughening and oxidizing, need to be investigated.

- *Helium Seal Data for Bolted Closures*—Seal design for the vessel main flange incorporates two Helicoflex O-rings located in grooves in the bottom flange face. Data are needed to demonstrate that the helium leakage requirement for the Vessel System can be achieved for normal operating temperatures and pressures.

Test facilities and testing machines required for tensile, impact, and creep testing of unirradiated materials are commonly available in the U.S. Irradiation facilities and hot cells are required for testing of irradiated materials.

It is anticipated that the testing can be accomplished within 2½ years. The cost estimates assume no cost for facilities and testing machines.

Table 8. Metallic vessel materials R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Irradiation Data for Reactor Vessel Materials											2,470
Properties of Heavy Section Vessel Materials											840
Reactor Vessel Emissivity											1,000
Helium Seal data for Bolted Closures											2,120
Total											6,430

3.2.3.3 Thermal Insulation. High-temperature fibrous insulation is used throughout the reactor system and the PCU notably in the hot duct, upper plenum shroud, Shutdown Cooling System (SCS) helium inlet plenum, and turbocompressor. The insulation is required to retain its resiliency and physical characteristics during normal operating and conduction cool-down accident conditions.

Data on the manufacture and performance of fibrous insulation are needed to ensure that the selected materials are capable of lasting for the life of the plant. The data include physical properties (heat resistance, heat conductivity, and heat capacity), mechanical strength at temperature, resistance to pressure drop, vibrations and acoustic loads, radiation resistance, corrosion resistance to moisture- and air-helium mixtures, stability to dust release and gas release, and manufacturing tolerances and mounting characteristics.

The acquisition of these data requires testing of insulation specimens or small assemblies of thermal insulation panels. Specific test rigs and facility requirements include helium flow, vibration, and acoustic test equipment as well as an irradiation facility and hot cell.

The testing to obtain thermal insulation data can be completed inside a two-year period. The total cost of these tests is estimated at $4.0 million assuming no cost for irradiation and hot cell facilities.

3.2.4 Graphite Materials and Ceramics

The graphite components of the reactor are the permanent side reflector, the core, and the core supports. The core includes the fuel elements and the replaceable reflector element. The reference material for the core, permanent side reflectors, and the core support is H-451 graphite. The reference material for the permanent side reflector support blocks at the hot duct entrance and selected core support post blocks is a purified form of HLM-grade graphite. The carbon/carbon (C/C) composite is proposed for the several subcomponents in the control rod assembly.

The graphite core support assembly includes hard ceramic pads that are located beneath the graphite elements. The ceramic pads thermally insulate the underlying metallic core support floor from the hot helium gas in the core exit plenum. The reference material is a commercially available aluminosilicate ceramic, grade AD-85.

Both H-451 and HLM graphite grades were used successfully at Fort St. Vrain. Additional data are required for the MHR to satisfy performance at higher temperatures and more stringent Quality Assurance (QA) requirements. The C/C composite and ceramic materials are relatively new reactor materials for which material properties are needed.

- *Graphite Multiaxial Strength Data*—The conceptual design of the graphite components has been performed based on the maximum stress failure theory. This theory is an approximation whose uncertainty needs to be quantified based on multiaxial strength data, and included in the ASME Subsection CE for permanent graphite core supports. The multiaxial strength data will also confirm the probabilistic structural design criteria for replaceable graphite core elements. Data are needed to determine the multiaxial strength surface of the core, core support, and permanent side reflector graphite. The database must be adequate to determine the mean value failure surface and the associated variability as well as the specified minimum strength surface.

- *Graphite Fatigue Data*—Fatigue analysis is required for the core graphite components. In this analysis, the fatigue strengths of graphite must be determined and the cumulative effects of varying stress amplitudes be accounted for. This includes both fatigue life for up to 10^5 cycles of uniaxial stress as a function of stress, and fatigue life when subjected to sequential series of uniaxial stress cycles with stress amplitudes of 65% to 100% of mean ultimate strength. These data are primarily needed for unirradiated graphite at room temperature. In addition, a limited number of data points are needed to determine the effects of temperature and irradiation effects, and the effects of oxidation on the fatigue properties.

- *Graphite Mechanical Properties Data*—The statistical variability of the mechanical properties of permanent and replaceable graphite components is needed for the development of probabilistic stress criteria. Data are needed to define the tensile and compressive strengths, elastic constants, and stress-strain relationship in accordance with appropriate ASTM standards for H-451 and HLM graphites, including the effects of orientation and location in billet, variation from billet to billet and lot to lot, temperature (ranging from shutdown condition to the maximum service temperature), fast neutron fluence (H-451 only), specimen size (volume), irradiation creep (H-451 only), and oxidation.

- *Graphite Irradiation-Induced Dimensional Change Data*—The statistical variability of the irradiation-induced shrinkage of core and core support component graphite is needed for the development of probabilistic stress criteria. Data are needed to define the irradiation-induced shrinkage and the associated variabilities for graphite H-451 as a function of fluence and temperature, including dependence on orientation and location in billet, variation from billet to billet and lot to lot, dimensional change covering a range of reactor temperature conditions, and dependence on the state of oxidation of the graphite.

- *Graphite Irradiation-Induced Creep Data*—The statistical variability of the irradiation creep properties of the replaceable core and core support graphite is needed for the development of probabilistic design criteria. Data are needed for H-451 graphite as a function of fluence and temperature, steady-state creep strain in tension and compression for up to 1% creep strain, transient (primary) creep strain, transverse-to-longitudinal strain ratios, creep under cyclic conditions covering a range of temperatures, and creep with stress reversal from compression to tension. In addition to the statistical database, data are also needed to establish the effect of creep on tensile strength, Young's modulus, thermal expansivity, and thermal conductivity. Furthermore, data are needed to validate that the creep strain is not significantly affected by the flux level.

- *Graphite Thermal Properties Data*—The statistical variability of the thermal properties of core and core support graphite is needed to develop probabilistically-based stress criteria. Thermal expansivity, conductivity, emissivity, and specific heat are needed for graphite H-451 and HLM, including dependence on orientation and location in billet, variation from billet to billet and from lot to lot, temperature dependence, dependence on neutron fluence and irradiation temperature, effects of oxidation, and effects of thermal annealing on thermal properties (during transients on irradiated H-451 only).

- *Graphite Fracture Mechanics Data*—The probability of functional damage to core graphite components needs to be assessed to ensure that the plant availability goal and the safety reliability requirements are met. Functional damage has been defined as a crack extending all the way across a fuel or reflector element or at least a significant distance into the element. So far, only vertical cracks have been addressed using existing continuum mechanics methods. Fracture mechanics methods are needed to address horizontal cracks and to validate the continuum mechanics methods. A database is needed to define the critical stress intensity factors and strain energy release rates for crack initiation, stable crack growth, and crack arrest for graphite H-451 at room temperature in air, including dependence on orientation and location in the billet and variation from billet to billet and from lot to lot. Additional data are also needed to establish the effects of the operating environment on the fracture mechanics properties including irradiation, temperature within the service temperature range, and oxidation.

- *Graphite Corrosion Data*—The graphite core and core support components may be corroded by coolant impurities, principally H_2O, with consequent deterioration of their integrity. Correlations describing the corrosion of H-451 graphite by coolant impurities during normal operation and H_2O ingress events are needed. Data are needed to characterize both the transport of coolant impurities and graphite corrosion products in H-451 graphite and intrinsic kinetics for the reaction of H_2O with H-451 graphite. To characterize the transport of coolant impurities in graphite, the porosity, tortuosity, and permeability of the graphite must be determined. To characterize the reaction kinetics, the reaction rate must be determined as a function of temperature, impurity concentrations, system pressure, and time.

- *Graphite Corrosion Data for Methods Validation*—The design methods and codes used to predict the extent of corrosion of graphite components by coolant impurities must be validated for normal operating conditions and for moisture ingress events. Particular attention must be given to transport of coolant impurities in fuel element graphite and to the effect of catalysis by graphite impurities and fission metals.

- *Graphite Destructive and Nondestructive Examination Data*—Destructive and nondestructive examination (NDE) techniques are needed for product control during procurement of graphite for the core and core support components. The NDE techniques need to be validated and test acceptance methods written in material procurement specifications for the procurement of graphite for core and core support components.

- *Graphite Coke Source Qualification Data*—Cokes produced from new feedstocks need to be qualified. Pre- and post-irradiation data are needed to qualify the candidate coke(s) for later use in preproduction and production graphite. Data must cover sufficient properties to ensure that the candidate coke reproduces known H-451 graphite behavior. These data include dimensional changes, electrical resistivity, elastic and sheer modulus, Poisson's ratio, sonic attenuation, strength, and coefficient of thermal expansion.

- *Graphite Oxidation Data for Postulated Accidents*—Computer codes used for analyzing air ingress during the heat-up portion of a conduction cooldown event need to be validated. Specifically, data are needed for air-graphite oxidation rates under low-frequency accident conditions that involve buoyancy-driven airflow and oxidation effects on graphite mechanical strength. Gas temperatures and composition are also needed to confirm graphite mass-loss rates and CO combustion models, and to assess the potential for flammable gas buildup.

- *Properties of High Temperature Control Rod Materials*—C/C composite is a candidate material for high-temperature control rods. An expanded property database for this type of material is needed, especially for depressurized conduction cool-down events, when control rod design temperatures reach nearly 1400°C. All basic material properties are needed including elastic moduli, tensile strength, low-cycle fatigue strengths, friction, and wear characteristics, as well as irradiation effects on all of the above properties. Additionally, fabricating characteristics, including machinability and weldability, are needed.

- *Hard Ceramic Insulation Properties Data*—The aluminosilicate ceramic insulation blocks used under the graphite core support structure are brittle and are susceptible to failures from fabrication imperfections as well as from the imposed working loads. To calculate the metallic core support and ceramic temperatures and the thermal stresses, a database is needed for flexural strength, compressive modulus, fracture toughness, thermal conductivity, and the effect of the reactor environment under nominal and off-nominal reactor operating conditions. The database must include dependence on orientation and location in billet, and variation from billet to billet, as well as grain size effect.

All tests related to unirradiated graphite material properties can be performed with standard test equipment. Irradiated property tests require an irradiation test facility and hot cell. The total cost of these tests does not include the cost for testing machines or irradiation and hot cell facilities. The graphite testing can be completed within three years, including screening tests on new graphite stock.

3.2.5 Component Development

Component design verification includes testing of scale models and assemblies, and demonstration testing of prototypical components. Major plant component verification testing is performed for reactor internals and hot duct, neutron control components, SCS circulator, SCS heat exchanger, turbogenerator, recuperator, precooler/intercooler, RCCS, fuel handling equipment, reactor service equipment, safety instrumentation, and an integrated assembly of the PCU.

Table 9. Graphite materials and ceramics R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Graphite Coke Source Qualification Tests					■	■					2,000
Graphite Mechanical Properties Tests	■	■									2,860
Graphite Irradiation-induced Dimensional Change Tests	■	■									5,810
Graphite/Air Oxidation Tests	■	■									1,860
Graphite Destructive and Non-destructive tests	■										540
Carbon/Carbon Composite Material Properties Tests					■						770
Hard Ceramics Insulation Properties Tests					■	■					1,030
Total											14,870

3.2.5.1 Reactor Internals and Hot Duct. The reactor internals and hot duct components include the graphite core and core supports, control rods, hot duct, and hot duct thermal barrier. Component test data are required to validate analytical performance predictions and confirm that the components will function without failure for the duration of their service lives. Specific data needs are:

- Graphite core support structure ultimate load capacity
- Thermal barrier response to the acoustically induced vibration environment
- Core column vibratory characteristics to assess potential effects of vibrations on core cooling
- Demonstrate no significant flow-induced vibrations that could inhibit rod insertion under reactor operating conditions with and without crossflow
- Failure load and stiffness data of H-451 core elements subjected to dynamically applied forces.
- Contact time and coefficient of restitution for input to analytical models for the prediction of seismic impact loads
- Failure loads and failure modes for replaceable fuel elements subjected to the combination of statically-applied mechanical and thermal loads, laterally applied mechanical loads at crack initiation, location of the crack, mechanical load at ultimate failure, and crack path from initiation to ultimate failure.

Large-scale flow, heating, vibration, and structural loading facilities are needed for control rod and control rod assembly tests, core column vibration tests, fuel element failure tests, fibrous insulation material tests, and hot duct vibration and acoustic tests. Such facilities exist at many laboratories in the United States.

3.2.5.2 Neutron Control Component. The GT-MHR neutron control system (NCS) controls reactivity within the reactor and must shut the reactor down upon command with high reliability. The NCS includes in-core flux mapping unit (IFMU) drives and ex-core fission chambers, control rod drives

(CRD), reserve shutdown control equipment, and microprocessor-based controllers to monitor neutron flux measurement and regulate control rod positions. Component test data are required to verify the design. Specific data needs are:

- Control rod control system performance characteristics (position versus time) and system response time. Reliability and performance data for all electromechanical components such as seismic response, operating speeds, effects of thermal aging, vibration and wear, accuracy and strength for all operating conditions.

- CRD reliability, speed of rod motion under normal and rod trip conditions, accuracy of rod positioning, and strength of the assembly under plant design and accident load conditions.

- Reactor Safety System operational performance data on the release mechanism, linkage and gate valve operation controls, pellet flow, and channel configuration to establish response time, material flow rates, material insertion time, and system reliability.

- IFMU detector and thermocouple signal data, including repeatability, linearity, drift, and signal noise levels.

- Guide tube frequency and magnitude of significant vibrations of guide tubes, plenum elements, and related components.

Tests of neutron system components require a high-bay facility for mounting a prototype control rod assembly in a vertical position. An autoclave is required to simulate reactor primary coolant conditions.

3.2.5.3 SCS Circulator. The MHR SCS circulator consists of a submerged radial flow compressor with integral electric motor, and a rotor system supported by electromagnetic bearings. The SCS circulator is housed in the reactor vessel.

To ensure that the SCS circulator performs as predicted, verification testing is needed for the impeller, shutdown loop shutoff valve (SLSV), and rotor assembly magnetic and catcher bearings. The specific data needs are:

- SCS circulator magnetic and catcher bearing tests to demonstrate the ability to withstand the impact from a circulator drop and coastdown

- SCS circulator prototype impeller aerodynamic and acoustic test data to optimize design and to demonstrate adequate primary coolant circulation for the various plant operating conditions

- Verification of the magnetic bearings and their controls, the variable speed induction motor, the motor cooling system, the circulator impeller and diffuser, and the SLSV under simulated reactor conditions.

New test facilities are required for the SCS circulator component tests, including bearing tests and aerodynamic and acoustic tests in an air flow facility. In addition, a high-pressure test facility (HPTF) is required for prototype SCS circulator unit tests in helium at reactor operating temperatures and pressure.

3.2.5.4 SCS Heat Exchanger. The SCS heat exchanger (SCS-HX) is a helium-to-water helical coil modular heat exchanger that is assembled from coaxially multistart coils. The principal function of the SCS-HX is to provide decay heat removal when the power conversion system is unavailable to perform this function.

The basic need for the GT-MHR SCS-HX is to reduce the uncertainty in the design. This requires understanding the flow distribution into the heat exchanger, assessing shroud seal leakage, and determining acoustic and flow-induced vibration loads on insulation cover plates and attachments. The specific data needs are:

- Obtain data on thermal cycling of fibrous insulation, effects of mechanical and acoustic vibrations, and effects of flow and thermal gradients
- Obtain SCS-HX tube vibrational fretting wear and sliding wear data and develop a wear protection method for the heat exchanger design
- Demonstrate ability to deliver and recover an NDE probe the full length of the SCS-HX tubes and determine the sensitivity of that equipment to detect tubing flaws
- Verify SCS-HX shroud seal design and determine leakage rate
- Determine frequency spectra and sound pressure levels generated by the SCS-HX helical tube bundle as a function of flow velocities and geometry variations
- Determine SCS-HX inlet flow distribution and magnitude of hot/cold streaks.
- Determine local heat transfer coefficients or “hot spot” factors and tube bundle flow resistance
- Demonstrate feasibility of coiling and threading multiple SCS-HX bare tubes in concentric coils. Determine thermal movements and interaction of tubes and supports.

A test facility needs to be constructed to accommodate a mockup of an actual heat exchanger bundle with shrouds. The air flow test rig used for the SCS circulator tests can also be hooked up to the heat exchanger.

3.2.5.5 Reactor Cavity Cooling System. The RCCS transports the core residual and decay heat from cooling panels in the reactor cavity to heat pipes in a natural draft stack. The primary function is the passive removal of the heat emitted by the reactor vessel during a conduction cooldown event.

The RCCS must ensure that fuel, reactor vessel, and structural concrete temperatures are maintained within allowable limits during reactor normal operation and passive cooling events. The specific data needs for the RCCS are:

- Metal surface emissivities under representative operating temperatures and surface conditions for all of the materials in the radiation heat transfer flow path
- RCCS natural draft stack flow and heat transfer to validate analytical models and verify the design for a range of heat loads and wind conditions
- Cooling panel heat transfer and friction factors to validate analytical models, including the effect of tube temperatures and heat fluxes, Reynolds number, riser internal surface conditions, and entry region condition

- Fuel block effective conductivity to verify values used in the RCCS heat removal calculation models

- Gas flow data to validate the computer models used to predict the heat transfer and gas mixing in the space between the core barrel and shroud and between shroud and reactor vessel wall, as well as the escape of hot gases to the upper reactor cavities.

A test rig for a full-scale partial cooling panel with a heat source to simulate decay heat is required as well as a wind tunnel facility for model testing of the natural draft stack. Emissivity testing only requires simple test apparatus.

3.2.5.6 Fuel Handling Equipment. The fuel handling machine (FHM) is an automated set of computer-controlled machines consisting of the fuel handling transfer mechanisms, fuel transfer cask, fuel handling equipment support structures, remotely-operated equipment positioners, hoist and grapple assembly, and fuel sealing and inspection equipment. Operation of the FHM is a key factor contributing to the plant availability. The system must be highly reliable with sufficient redundancy to accommodate upset conditions and equipment failures.

Fuel handling equipment improvements since Fort St. Vrain need to be demonstrated to show that operations are performed safely and reliably within FHM cycle time allocations. These include grapple operation, viewing, and electronic control. Data are principally needed to establish the functional and performance limits of components under expected environmental conditions. The specific data needs for the fuel handling equipment are:

- Grapple system and remote services connections operability and reliability, as well as functional and performance limits under the expected environmental conditions

- Vertical drive system operability and reliability for the grapples and the positioning capability of the overhead crane, as well as functional and performance limits under the expected environmental conditions

- Performance characteristics of instrumentation and control components, including limiting values for element motion, direction, velocity, size of identification marking, and temperature

- Verification of fuel handling system physical compatibility, alignment requirements, tolerances, and coordination by the control system, as well as human factors data on the control station.

- Position certainty of the fuel handling positioner control axes under expected static and dynamic load conditions

- Verification of fuel handling equipment valve and seal leakage, with and without load of supported equipment and with misalignment of neutron control assembly housings

- Verification of fuel sealing and inspection equipment automated packaging, sealing, and inspection process, including extended cycle endurance prior to shipment and installation in the plant.

The development of the fuel handling equipment will be accomplished by a series of individual component and assembly tests. No large-scale facility construction is assumed to be required.

3.2.5.7 Reactor Service Equipment. The Reactor Service Equipment (RSE) consists of remotely operated shielding casks and mechanisms that are used for inspecting and maintaining reactor, power conversion, and fuel handling equipment components. These include tools for viewing, grappling, and removing broken core components, handling of components taken to the hot cell for inspection, repair and testing, removing reserve shutdown material from the core, retrieving and installing in-core neutron detectors, retrieving and installing the power conversion unit generator and turbomachine, and inspecting and surveying reactor components in place.

The design of most RSE components for the MHR is based on previous designs developed and proven at Fort St. Vrain. Data needs for these designs are only to verify performance and reliability of the service equipment hardware. The specific data needs are:

- Reactor service equipment tolerances and compatibility with operating environment, as well as data on the capability of the tool to perform the required work in a timely manner
- Acceptable installation and mechanical operation of in-service inspection and surveillance equipment, acceptable image transfer, and acceptable removal and transfer of material surveillance samples
- Generator and turbomachine service equipment handling and movement capabilities to perform removal and installation operations with required accuracy.

Performance testing of the service tools requires a high-bay area with capacity to handle heavy equipment.

3.2.5.8 Safety Instrumentation. The MHR safety instrumentation and controls include measurements used for protection, monitoring, and control. These include mass flow, temperatures and fission product plate-out, and neutron detection. In general, the plant design goal is to utilize commercially proven controls and instrumentation, thereby minimizing the need for additional instrumentation and controls development. The development necessary is limited to verifying the design and operation of this equipment to ensure safe, reliable, and repeatable component operation. The specific data needs are:

- Verification of helium mass flow measurement at the turbine exit for measurement of core flow with respect to flow instrumentation static and dynamic performance at plant service conditions
- Verification of temperature instrumentation with respect to capability of monitoring core and vessel temperatures during conduction cool-down events and accuracy of measurement
- Verification of plate-out probe accuracy of detecting condensable activity in a high-temperature helium environment
- Confirmation of neutron detector and cabling integrity, operability, neutron sensitivity, and detector/cable life at in-reactor conditions.

Neutron detector and cabling tests require a nuclear test reactor and test furnace to test neutron detector performance under simulated temperature and flux conditions. A hot helium flow test facility is required for helium mass flow tests and temperature probe tests. Plate-out probe tests can be performed in a helium radiochemistry laboratory.

3.2.5.9 Cost and Schedule. It is anticipated that all component data needs and testing can be accomplished within a four-year period. The cost estimates include all test labor, hardware, and the costs of new test facilities, where identified.

Table 10. PMR component development R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Reactor Internals and Hot Duct											11,720
NCA Components											3,740
SCS Circulator											8,500
SCS Heat Exchanger											2,500
Reactor Cavity Cooling System											2,950
Fuel Handling Equipment											6,500
Reactor Service Equipment											4,660
Safety Instrumentation and Control											4,000
Total											44.570

3.2.6 GT-MHR Component Development

Much of the MHR component development, such as the power conversion equipment, is specific to the gas turbine concept, including the turbomachine and heat exchangers.

3.2.6.1 Turbomachine. The turbomachine consists of the helium turbocompressor and generator housed in the power conversion vessel unit. The turbocompressor consists of a two-section compressor (separated to facilitate intercooling) and turbine. The turbocompressor drives the synchronous submerged generator. The vertical turbomachine rotor assembly is supported on an active, magnetic bearing system and catcher bearings. The specific data needs are:

- Demonstrate journal and thrust catcher bearing performance, electric control system performance, and rotor dynamic stability with prototypical bearings,
- Determine electrical properties of generator windings in a helium environment and obtain insulation data under simulated blow-down conditions to confirm structural integrity.
- Demonstrate the ability of the turbocompressor casing to contain missiles as a result of turbine deblading.
- Determine turbine rotor vibration characteristics, including rotor natural frequencies and deflection magnitudes.
- Determine static seal system (e.g., seal between the turbocompressor and inlet ducts) performance and determine materials data for segmented piston seal rings and the mating surfaces with which the seals are in contact. Obtain data on seal coating materials, life expectancy of materials as a function of wear, and the coefficient of friction in helium to be used for resistance to sliding motion.
- Assess fission product plateout (e.g., silver, tellurium) on turbine blades, which may cause strength degradation and require use of protective coating, and determine the need for oxide coating to avoid galling or self welding as a result of rubbing of parts.

- Determine flow velocity and temperature profile at the turbine inlet to quantify potential flow maldistribution.

- Validate analytical performance maps by obtaining the precise configuration of compressor blades and turbine blades derived from experiments.

New test rigs are required for many of the turbomachine tests, such as flow distribution tests, bearing tests, system seal tests, and rotor dynamics tests. Tests in helium environment are limited to material samples and full or subscale components requiring limited volumes of helium. Facilities for testing electrical insulation materials are in existence.

3.2.6.2 Recuperator. The PCU recuperator is a modular counterflow, gas-to-gas, plate-corrugated surface heat exchanger in a parallel arrangement. The material of construction for all parts of the recuperator is austenitic stainless steel. The recuperator is considered state-of-the-art technology, thus only limited development is needed to establish primary coolant flow distribution, confirm thermal performance, and develop methods for leak detection. The specific data needs are:

- Determine recuperator inlet/outlet flow distribution between modules in parallel to confirm heat exchanger effectiveness

- Determine recuperator friction factor and heat transfer coefficients as a function of Reynold's number

- Develop recuperator leak detection methods and equipment.

All recuperator tests can be performed in existing airflow test facilities.

3.2.6.3 Precooler/Intercooler. The PCU precooler (PC) and intercooler (IC) are modular helium-to-water, once-through, straight-tube heat exchangers with outer fins on the gas side. Helium flows on the outside of the tubes counterflow to the water on the inside of the tubes. The tube material is austenitic stainless steel. The specific data needs are:

- Determine the flow distribution and magnitude of hot/cold streaks at various cross sections, including the inlet to the PC/IC tube bundles

- Determine leak rates for PC/IC high pressure seal arrangement to assess coolant bypass

- Determine presence of PC/IC flow-induced vibration characteristics, such as flow-induced turbulent buffeting, vortex shedding, or fluid-elastic instability, which can cause dynamic instability and tube damage

- Confirm inspection capability and inspection equipment sensitivity for the specific PC/IC tube circuit geometry

- Confirm PC/IC shell- and tube-side heat transfer characteristics and shell-side flow resistance, and determine the effective flow resistance of the finned tube bundle

- Quantify the tubeside erosion/corrosion rates as a function of the operating parameters, water chemistry ranges, and tube geometry.

All PC/IC tests can be performed in existing air flow test facilities.

3.2.6.4 Power Conversion System Integrated Test. The PCU of a GT-MHR couples a closed-cycle helium gas turbine to the high-temperature GCR. An integrated test of the PCU is required prior to operation of the first module to confirm the performance of the prototype components under nominal and transient plant conditions, and their sustained operation in the nuclear environment. The test will determine the integrated performance of each component (e.g., thermal efficiency and pressure drop), check local deformations caused by thermal interactions, measure helium bypass flows, verify the control and protection system response to design transients, and verify inspection and component replacement procedures.

It is anticipated that all component data needs and testing can be accomplished within a four-year period. Most turbogenerator subcomponent and separate effect tests will be completed early so that the data generated can be factored into the turbomachine final design and proof testing in the PCU integrated test facility.

3.2.6.5 Cost and Schedule. The cost estimates include all test labor, hardware, and new subcomponent test facilities, as needed. The cost of the PCU integrated test is based on the design of a separate test facility with a closed loop combustion gas turbine providing the required test conditions. The cost does not include prototype component cost or facility costs, but includes the cost of facility operation and maintenance. The test facility design phase is 24 months followed by 12 months for construction and PCU module installation, and 12 months of integrated testing.

Table 11. GT-MHR component development R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Turbomachine											25,600
Recuperator											3,100
Precooler/Intercooler											4,350
PCU Integrated Test											29,600
Total											62,650

3.3 Very-High-Temperature Reactor

The basic technology for the VHTR has been established in former HTR plants (DRAGON, Peach Bottom, AVR, THTR, Fort St. Vrain). In addition, the technologies for VHTR are being advanced in the Near-Term Deployment project on GT-MHR and PBMR and in the International Near-Term Deployment project on PBR and PMR. Furthermore, complementary R&D is necessary to enable the system for an increase in operating temperature beyond 850–900°C and to develop the interface between the Nuclear Heat Supply System (NHSS) and the heat utilization systems.

The HTTR and HTR-10 projects will demonstrate the feasibility of VHTR as well as co-generation/nonelectric applications. Additionally, these projects will provide first data from nuclear operation of VHTR in addition to the results from the former nuclear process heat (NPH) projects in Japan and Germany of Prototype Nuclear Process Heat (PNP), which are still of high relevance for future VHTR development.

The increase in temperature is a general challenge to improve the thermal efficiencies of any power conversion system for electricity generation or any other process heat application, like thermochemical water-splitting, refining/up-grading of crude oil, petrochemical processes, steam reforming of methane, aluminum oxide production, etc. Most of these applications need specific component and process developments for adaptation to the NHSS, which can be largely standardized.

Nuclear process heat applications have to compete with highly developed conventional production processes and very efficient combined heat and power (CHP) plants as used in industry. In addition, the safety requirements will be more challenging compared to dedicated electricity generation at remote sites due to the fact that NPH plants could be expected to operate in an industrial environment with high population densities and large investments for the production facilities. In contrast to electricity production, an NPH plant represents only an auxiliary facility to provide energy. Therefore, maximum permissible operational and accidental releases might be much less than current licensing limits. Any interruption of the production may cause more financial damage than for a plant producing electricity as part of an integrated grid. Thus, reliability requirements will be very high and can only be coped with by multi-unit modular reactors without long outages. Contamination of the product will have to be precluded to make the product acceptable in the market for consumers critical about low additional radiation. On the other hand, the nuclear island will have to be sheltered against external impacts, like gas or pressure vessel explosions, as well as fire from flammable products. This might, for example, require below-grade siting. Ease of decommissioning after operation will also be indispensable.

It is expected that the industrial customer requirements for CHP and for special applications like nuclear steam reforming on a refinery site will have to be identified as a first step by creating joint working groups between the nuclear suppliers and potential NPH users. In addition, information from former NPH projects needs to be retrieved and archived for future use.

Intermediate heat exchangers and hot gas isolation valves will be necessary for radiological separation of the nuclear island and the production facilities despite the rather clean primary circuit of VHTR. This is especially the case for isotopes like tritium, which can easily permeate metallic barriers at high temperatures. Special coatings need to be developed to reduce this effect. The development of compact and reliable IHX as well as hot gas valves will benefit from the development of the recuperator and by-pass valves for direct cycle plants.

Process steam for many petrochemical and chemical processes will require a contaminant-free steam cycle and possibly a combined cycle VHTR design. The financial drawback of an intermediate cycle for CHP purposes could be expected to be compensated by the higher efficiencies of combined gas and steam cycle turbines, which could reach more than 50% efficiency, and by the use of conventional power generation or process engineering equipment in the secondary loop behind the isolation valves. Hot gas ducts, IHX, and valves are the key components for NPH and will require testing under more stringent transient and accident conditions in addition to the ongoing operation of such components in the HTTR. These tests could be performed, for example, in the helium test loops that will be available after 2006 in France.

The steam reformer is a key component for hydrogen production and petrochemical applications and still has to be optimised for use in an intermediate HTR circuit. This requires the qualification of high-temperature alloys (and coatings) that are also resistant to corrosive gases, such as hydrogen, carbon monoxide, and methane, and the development of new catalysts capable of being easily exchanged. Collaboration with the HTTR project should be established and performed by the Generation IV International Forum side (e.g., by concentrating on safety aspects of steam reformers such as transient analysis, fatigue effects, tritium permeation, gas explosions, etc.).

Other specific components for petrochemical or metallurgical use should be designed in a first step in consultation with process engineering companies to facilitate identification of further R&D issues for a broader application of NPH. Many thermochemical processes have been evaluated by GA in the United

States from various viewpoints, such as a number of chemical reactions, existing data, corrosion problems and so on. It was found that the thermochemical iodine-sulfur (IS) process is the most promising water splitting method for HTGRs.

In summary, R&D issues for VHTR are complementary to those for PBR and PMR by extension of temperatures, efficiencies, and applications beyond dedicated electricity generation, and should be performed in parallel to make best use of potential synergies between these R&D activities (see Figure 6).

The main R&D issues for VHTR are discussed below under the assumption that the basic R&D for PBR and PMR will be performed separately. Close correlation between these activities should be assured via the Generation IV International Forum.

3.3.1 Fuel

The present HTR fuel was developed about 20 years ago for large-sized HTRs, and the fuel was only optimized for low fission product release for the intended operational conditions (750–850°C; ~6 MW/m^3). The TRISO particle was a result of former direct-cycle plants to allow for easy maintenance of the gas turbine within the primary circuit. Later on, it was discovered that this type of fuel is capable of fission product retention up to more than 1600°C where chemical degradation of the SiC layer starts. This feature gave rise to the concept of the modular HTR (MHR) with inherent limitation of maximum accident temperatures below this limit. On the other hand, the TRISO particle was not yet optimized for application in MHR or for retaining these features under higher burn-up conditions.

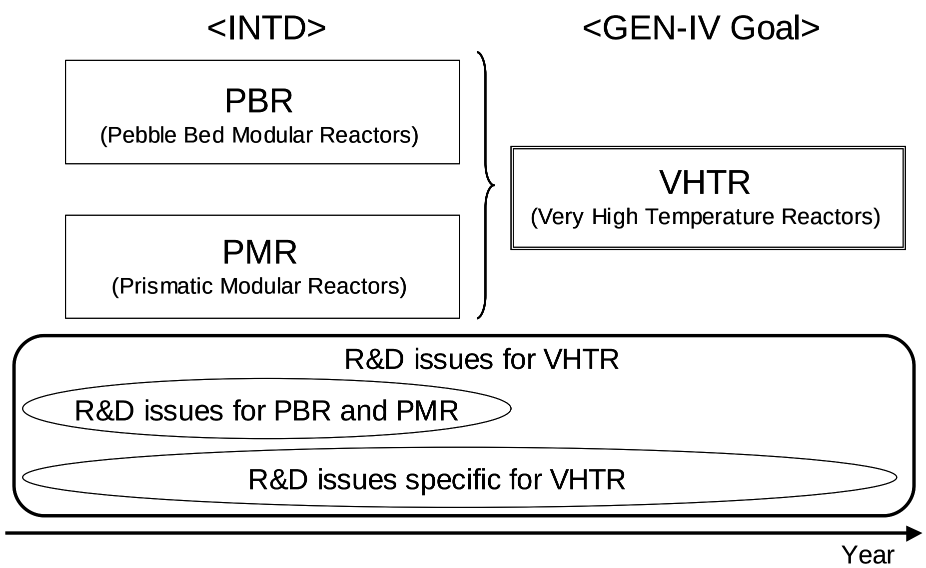


Figure 6. Complementary and synergetic R&D of PBR, PMR, and VHTR.

The main targets and motivations for VHTR fuel development are:

- Increase in operational coolant temperatures from 850°C up to 950–1100°C
- Increase in tolerable accident temperature limits beyond 1600°C
- Increase in maximum burn-up from 80 GWd/t up to 150–200 GWd/t

- Higher power density above $6MW/m^3$.

In a second, more future, development phase, the reactor exit coolant temperature could be targeted toward 1300°C, if fully ceramic or ultra-high-temperature gas turbines become available. These R&D issues are specific for VHTR because PBR and PMR stay within the present capabilities of the fuel. The present maximum value of reactor exit coolant temperature is 850°C in the Japanese HTTR, with a maximum temperature of 950°C anticipated in 2003.

The increase of helium coolant temperature at outlet of an HTGR results in an increase of fuel temperature. Accordingly, it is necessary to develop fuels that resist higher temperatures. Fuel particles coated with SiC are used in HTGRs (such as the GT-MHR, Japanese HTTR, Chinese HTR-10, and PBMR designed in South Africa) for a helium coolant temperature, at reactor outlet, of less than 950°C. For higher temperature utilization, materials with greater refractory capability than SiC are required.

The potential for ZrC utilization in HTGRs enables increased power density and an increase in power size under the same coolant outlet temperature to strengthen resistance against chemical attack by palladium for plutonium-burning HTGRs. Thus, such developments could also be important for PBR and PMR design improvements and extension of safety margins. On the other hand, additional R&D for fuel development and fuel qualification, or design study on countermeasures on the plant level is needed because the ZrC is easily oxidized. Oxidation protection layers for the fuel elements could be another approach to cope with this drawback.

3.3.1.1 SiC. The following SiC-coated fuel particle R&D activities will be performed in the PMR (see 3.2.1), and are important precursor activities for follow-on VHTR development.

- Demonstration test fuel fabrication and irradiation ($1.0 million)
- Fuel fabrication process improvement ($20.3 million)
- Fuel qualification via irradiation tests, post irradiation heat up tests, etc. ($29.5 million)
- Radio nuclide transports ($17.9 million).

3.3.1.2 ZrC. As shown in Figures 7 and 8, a ZrC fabrication method was established at JAERI's Tokai Research Establishment using the Bromide process with the 50g-scale coater (Ogawa, et al. 1981). Commercial-scale will be around 3 kg-scale.

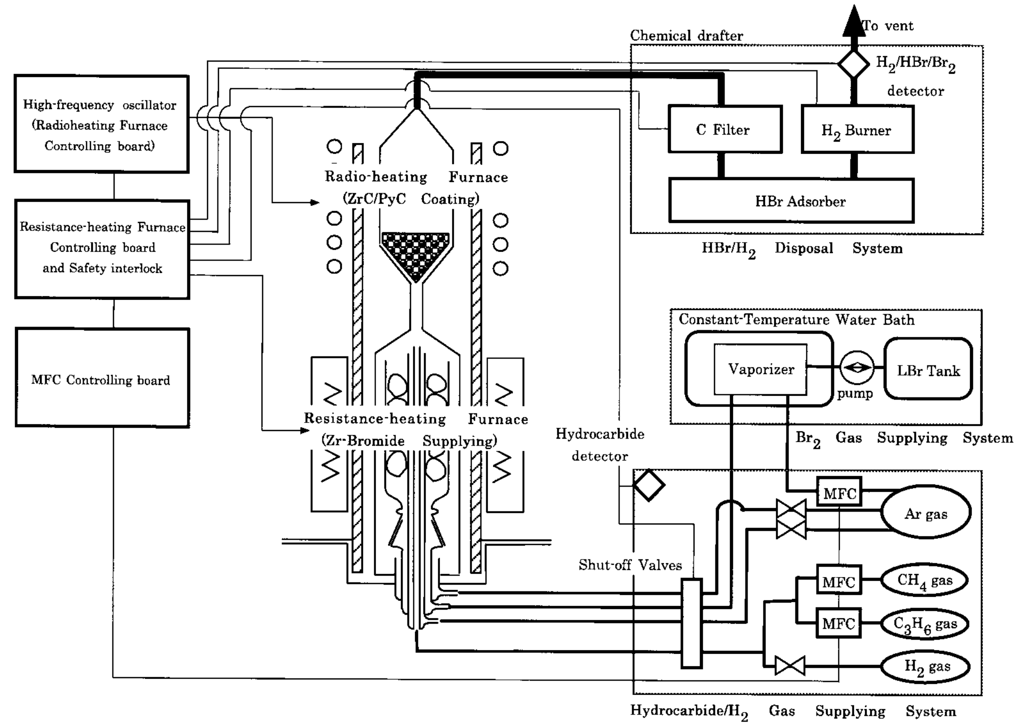


Figure 7. Flow diagram of ZrC-coater in JAERI.

Figure 8. Electrical heating furnace of ZrC-coater in JAERI.

Irradiation tests were carried out up to a burnup of 4.5% FIMA and a fuel temperature of 1300–1500°C. As shown here, ZrC-coated fuel particles demonstrated improved irradiation performance (both in maximum fuel temperature and maximum burnup) than SiC-coated particles (Ogawa, et al. 1992; Minato, et al. 2000).

Item	SiC-coated fuel particle	ZrC-coated fuel particle
Maximum fuel temperature	< 1600°C	> 1800 ~ 2000°C
Maximum burnup	~100GWd/t	100 ~ 200GWd/t

For commercial application, burnup of ZrC-coated fuel particles should be extended to more than 10% FIMA. R&D efforts for ZrC-coated particles are as follows:

1. Demonstration tests for commercial-scale (>3 kg batch) fabrication:
    a. *Status*—Commercial-scale (>3 kg batch) fabrication should be demonstrated.
    b. *Time duration*—5 years.
    c. *Total R&D cost*—$0.1 million for 0.1 kg batch-scale test and $1.5 million for commercial-scale test, including coater fabrication.
    d. *Technological difficulty*—Optimization of deposition condition to obtain stoichiometric ZrC is needed for commercial-scale coater.
2. Irradiation test of ZrC:
    a. *Status*—Irradiation tests should be carried out up to 10% FIMA.
    b. *Time duration*—5 to 10 years.
    c. *Total R&D cost*—including postirradiation tests; $3M, not including cost of reactor operation.
    d. *Technological difficulty*—ZrC behavior should be examined, including grain/crystal growth under high burnup.
3. Measurement test of diffusion coefficients:
    a. *Status*—Diffusion coefficient of metallic fission product, especially silver, should be investigated through irradiation and postirradiation tests.
    b. *Time duration*— 5 to10 years, along with irradiation test of ZrC (see item 2 above).
    c. *Total R&D cost*—including postirradiation tests; $1+ million for postirradiation heating test.
    d. *Technological difficulty*—None.
4. Study on oxidation resistance and corrosion protection layers:
    a. *Status*—Countermeasures against ZrC-oxidation should be developed. First tests on coating of fuel pebbles have been successful but still need improvements in the coating technique for the fuel matrix material.

b. *Time duration*—5 years.

c. *Total R&D cost*—$1.0 million.

d. *Technological difficulty*—Some technical breakthrough is necessary to solve the problem on the coated-particle level. Oxidation protection layers on the fuel element have to be tested under irradiation and mechanical loads.

5. Development of failure model of the ZrC-coated fuel particle:

   a. *Status*—Failure model of the ZrC-coated fuel particle should be developed for fuel and safety design.

   b. *Time duration*—3 years.

   c. *Total R&D cost*—$0.5 million.

   d. *Technological difficulty*—Technical breakthrough will also be needed to develop the failure model of ZrC-coated fuel particles. However, based on experience developing the SiC-coated fuel particle model, a ZrC-coated particle model could be realized within 3 years after the irradiation data is obtained.

The R&D schedule and cost are shown in Table 12. (The R&D cost does not include personnel expenditure.)

Table 12. ZrC fuel R&D schedule and cost.

Year R&D	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	R&D Cost(k$)
(1)																	6,500
(2)																	69,000
(3)																	27,500
(4)																	1,500
(5)																	500
Total																	105,000

Ultra High Burn-up and Transmutation Fuel. New fuel cycles for ultra-high burn-up or for transmutation of minor actinides need specific fuel development, which will also affect the composition of the kernel and the fabrication methods, which must be remote if minor actinides are to be recycled. Due to the formation of CO, oxide fuel might not be advantageous for ultra-high burn-up. Other stoichiometric compositions, like oxicarbides (e.g., UCO or UC), should be investigated in this case. UCO R&D is mainly described in the preceding section on PMR, but needs to be extended to Pu/Np driver fuel and minor actinide-based fuel transmutation as a next step.

1.	Driver Fuel Development	Remote fabrication and QA, suppression of Ag release/Pd attack, feedback from fuel disposal behavior, modeling of driver fuel, optimization with regard to disposal/reprocessing
2.	Transmuter Fuel Development	Remote fabrication and QA, optimization of safety and disposal behavior, creation of a technical basis for transmuter-fuel fabrication, modeling of transmuter fuel behavior, radiotoxicity analysis

3.	Reprocessing / Repackaging	Recovery of existing knowledge; R&D on proliferation-resistant processes, separation of kernels, coatings and matrix; minimization of secondary waste and of gaseous/liquid effluents.

The cost for the total program cannot be fixed. However, with regard to the importance of these activities for sustainable deployment of HTR, it is recommended that an adequate share of the budget (e.g., $3-5 million/year) be invested over the next 15 years to support these R&D efforts.

3.3.2 Fuel Cycle

The minimization, management, and disposal of radioactive waste are key issues for the present and future use of nuclear energy. At present, the 125 GWe of nuclear power in the European Union produce about 2,500 tons of spent fuel annually, containing about 25 tons of plutonium; 3.5 tons of the minor actinides neptunium, americium and curium; and about 100 tons of fission products, of which 3.1 tons are long-lived fission products such as cesium, iodine and technetium.

Actual reprocessing of LWR fuel and a first recycling as MOX fuel in LWRs already contribute to a significant reduction of waste volumes and radiotoxicity in comparison to the once-through cycle. Nevertheless, the stockpiles of transuranic actinides are steadily increasing worldwide and urgently need improved and effective strategies for the back-end of the fuel cycle on national and supra-national levels. In addition, multi-recycling of MOX fuel in LWRs cannot easily be performed due to the degradation of fissile contents and of nuclear stability with respect to reactivity control.

Symbiotic LWR and HTR fuel cycles can achieve improved waste minimization performance because of the specific features of HTR core physics and the epi-thermal neutron spectrum. One of the benefits of HTRs is that they are able to accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design, as particularly demonstrated in the AVR test reactor in Germany. This flexibility is due to a decoupling between the parameters of cooling geometry and those that characterize neutronic optimization (e.g., moderation ratio or heavy nuclide concentration, self-shielding effects, distribution, etc.). In fact, it is possible to modify the packing fraction of coated particles in the fuel within the graphite matrix without changing the dimensions of the fuel elements. Other physical reasons favor the superior adaptability of HTRs with regard to the fuel cycle in comparison with reactors using moderators in the liquid form, such as LWRs or liquid metal reactors. An illustration of this adaptability is the void coefficient, which limits the plutonium content of PWR MOX fuels but is not a constraint for HTRs. It can also be noted that an HTR core has the potential for a better neutron economy than a LWR because there is much less parasitic capture in the moderator (the capture cross section of graphite is 100 times less than that of water). The much larger thermalization length leads to an epi-thermal neutron spectrum with considerable fast flux contributions. Finally, HTR fuels are expected to be able to reach very high burn-ups far beyond the possibilities offered by other thermal reactors. All these capabilities permit essentially complete plutonium fission in a single burnup and minimize the proliferation risk. The HTR fuel may also provide an ideal encapsulation for spent fuel that is potentially much more resistant to leaching in a final repository than vitrified waste. Therefore, The radiotoxicity of the residual waste from HTR would be much less, as long as the coating of the fuel particles is still intact.

Future R&D should identify innovative HTR fuel compositions for ultra-high burn-up of transuranic elements and wastes from LWRs (e.g., reprocessed spent MOX). Preliminary investigations show that a high-temperature, inherent-safe transmuter (HIT) can reduce the mass of radioactive waste from 100% down to about 30% by using special driver fuel containing Pu and Np together with transmutation fuel containing the other minor actinides. Further reductions down to 5% can be achieved by successive burning of the very same transmutation fuel in a high-temperature, accelerator-driven transmuter (HAT) without intermediate reprocessing steps. In the future, HTRs could even provide a further reduction of the waste in a fast spectrum. The HIT, HAT, and GFR systems also largely make use of the basic HTR technology with regard to the power conversion system, safety philosophy, fuel development, etc. This fact leads to an efficient use of R&D efforts on all types of GCRs.

The necessary design work and studies on the neutronics is estimated to require about $2 million per year over the next 5 years.

R&D on reprocessing and disposal of SiC-/ZrC-coated fuel particles is needed as follows:

- *Long-period repository/direct disposal for SiC-particles.* It is believed that the SiC-coated fuel particle acts as a miniature containment vessel to retain fission products during long-term repository or direct disposal. Confirmatory tests, which prove the long-term intactness of the coating layers, are needed as R&D.

- *Long-period repository/direct disposal for ZrC-particles.* The long-term behavior of the irradiated ZrC layer is unknown. Further, R&D for the irradiated ZrC-coated particle, such as oxidation rates under repository or direct disposal conditions long-term confirmatory tests, should also be performed.

- *Reprocessing procedures for SiC-coated fuel particle.* JAERI focused on the applicability of graphite-CO_2 reaction techniques and the jet grind method to head-end reprocessing (see Figure 9). The graphite-CO_2 reaction technique was developed to reduce CO gas release in the burning process of the fuel compacts and fuel kernel. The jet grind method was investigated to reduce maintenance work of roll-gap clearance of the roll grinder, which was used to remove the SiC layer from burned SiC-particles. These methods should be demonstrated in industrial-scale.

- *Reprocessing procedures for ZrC-coated fuel particle.* Since ZrC-coated fuel particle is easily oxidized, it is believed that the head-end process may be more straightforward compared to the process for SiC-coated particles. As such, no major R&D may be needed for the head-end process for ZrC-particles.

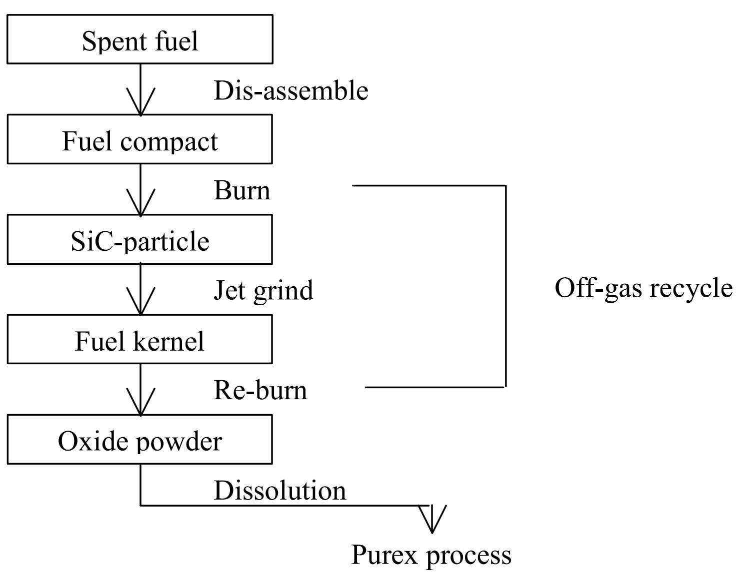


Figure 9. Block diagram of JAERI head-end reprocessing.

3.3.3 Reactor Systems

3.3.3.1 Metallic Materials. Metallic materials for IHX and high-temperature pipe can be used below 950°C for reactor outlet coolant temperature, but conventional materials can generally withstand temperatures above this. Accordingly, development of metallic superalloys and ceramic materials and coatings are expected for application in nuclear plants, as is the case in other high-temperature industries.

R&D of metallic materials below 950°C was completed in JAERI as follows:

1. Hastelloy XR
   a. Both base material and filler material for welding were developed (patented in Japan and the United States)
   b. Large amounts of engineering data were accumulated
      (1) Hastelloy XR was used for HTTR as a high-temperature structural material
      (2) Data were used for design and safety evaluation (tension, creep, fatigue, corrosion, etc.).

2. Ni-Cr-W superalloy (up to 1000°C)
   a. Both base material and filler material for welding were developed (patented in Japan and the U.S.)
   b. Small amounts of engineering data were accumulated.

3. Alloy 800H
   a. Engineering data were accumulated. Alloy 800H was used for HTTR as a cladding material for the control rods. Data were used for design and safety evaluation (mainly creep data for irradiated material).

4. 21/4Cr-1Mo Steel
   a. Engineering data were accumulated. 21/4Cr-1Mo steel was used for HTTR as pressure boundary components. Data were used for design and safety evaluation (Charpy impact, fracture toughness for aged/irradiated material, fatigue, etc.).

3.3.3.1.1 High Temperature Alloys—The development of high-temperature materials over the last 60 years has been largely driven by the aerospace industry, in particular the materials requirements for military and civil aircraft engines. Cast alloys capable of operation at temperatures around 950°C have been developed based on directional solidification and single crystal production methods. Using thermal barrier coatings and component cooling systems, it has been possible to increase the environment temperatures, such as the combustion temperature in a gas turbine, to 1200°C and even higher. Significant effort is currently being put into the transfer of the materials technology from the relatively small-scale application in an aerospace gas turbine to the much larger-scale stationary gas turbine for power generation.

For high-temperature reactor systems, many of the aerospace alloys will undoubtedly be evaluated and used. The stationary gas turbine will use the same materials as the conventional industrial gas turbines, after the appropriate qualification for nuclear applications. For heat exchangers and steam generators operating at very high temperatures, tube, pipe, and vessel materials are required. Apart from the materials qualified in the German and Japanese HTR projects in the 1980s, there are few new developments. The gradual maturing of the wrought, oxide dispersion strengthened alloys based on iron or nickel, may provide a class of materials capable of operating at temperatures up to 1100°C.

Beyond 1100°C, it is necessary to make a significant leap to refractory based metals, such as molybdenum and tungsten, which pose difficult challenges in the area of manufacture and application due to the extremely poor hot corrosion behavior in oxygen-containing atmospheres.

An open question is the behavior of the materials under irradiation conditions and the problem of activation due to the usually significant amounts of cobalt in these materials, which needs to be evaluated.

3.3.3.1.2 Vessel Materials—The development of 9% Cr steels has resulted in the availability of martensitic-ferritic materials, which offer high creep strength (similar to the austenitic steels, such as AISI 321 and 347) at temperatures up to 650°C. Such steels may find applications as pressure vessel steels in the modular HTR design and may avoid the necessity of vessel wall cooling. There may be problems in obtaining the required 100% martensitic microstructure in thick-walled components, and the propensity of weldments towards type 4 failure (soft heat affected zone) is a cause for concern. The irradiation behavior of this class of steels is, to some extent, known. Material tests of the tendons for prestressed cast-iron vessels are included in the prestressed cast-iron vessels (PCIV)-related R&D.

3.3.3.1.3 Coatings—The development of coating systems will be an important aspect as it becomes increasingly difficult to combine mechanical strength with the environmental properties such as hot corrosion and friction/wear. Major concerns are the failure modes and the prediction of useful life. Reactor-specific problems may also be amenable to solutions that employ coating systems, for example the prevention of tritium permeation through the component walls.

3.3.3.2 Graphite. New fine-grained isotropic graphite with high strength and low irradiation damage will be necessary to achieve high outlet-gas temperature as well as long life. Targets are high tensile strength (1.5 times higher tensile strength than that of IG-110) and almost the same thermo-mechanical properties as IG-110.

- Current status
  - Several properties (strength, thermal properties) without irradiation condition have been obtained
  - Irradiation testing is being planned.
- R&D issues
  - Establishment of design database (strength, thermal properties, etc.)
  - Development of material modeling under irradiation to predict property changes at high neutron fluence.

3.3.3.3 Ceramics.

3.3.3.3.1 Development of C/C Composite—Development of C/C composite is necessary for a control rod sheath for a prismatic-type VHTR because the core metallic material cannot withstand neutron irradiation and high temperatures above 1000°C.

- Current status

- Strength data as well as thermal data at real-time and elevated temperatures were obtained for 2D woven C/C composite
- Development of stress analysis code is ongoing
- Some irradiation test data has been obtained.

• R&D issues

- Material tests on mechanical properties, thermal properties, fracture behavior, etc., including oxidation effect and post irradiation to establish design guidelines and a design database
- Modeling of material behavior and stress analysis code considering anisotropy
- Obtaining nondestructive testing data and fracture toughness data to establish acceptance guidelines
- Trial manufacture and out-of-pile tests.

3.3.3.3.2 Superplastic Ceramics Application—Over 1000°C, ceramic materials are thought to be one of the more promising structural materials, although they have the disadvantaged of bonding, pressure-forming, and machining. By applying the superplastic ceramics, it is possible to conduct the bonding and pressure-forming in various structural shapes. Two types of superplastic ceramics are being investigated: low thermal conductivity (e.g., fine-grained zirconia ceramics, for an inner tube of coaxial double tube) and high thermal conductivity (e.g., fine-grained SiC ceramics, for heat transfer tubes in a heat exchanger).

• Current status

- Superplastic deformation on fine-grained zirconia ceramics are being studied; data for tensile, bending, and compressive deformation characteristics have been obtained
- Ion irradiation effects on Superplastic deformation for fine-grained zirconia ceramics are being studied; data of self ion, Zr ion, and irradiation have been obtained.

• R&D issues

- Obtain a database (superplastic deformation characteristics, residual stress after superplastic deformation, optimum forming temperature and strain rate, etc.) on pressure-forming technology by superplastic deformation
- Obtain a database (strength, residual stress, etc., after bonding process, optimum bonding temperature and pressure, etc.) on bonding technology by superplastic deformation
- Material tests on mechanical properties, thermal properties, fracture behavior, etc., including post irradiation to establish design guidelines and a design database
- Trial manufacture using bonding and pressure-forming technology, and out-of-pile tests to confirm the structural integrity
- Obtain nondestructive testing data and fracture toughness data to establish acceptance guidelines.

3.3.3.4 Development of Prestressed Cast-iron Pressure Vessels. Nuclear reactor design is strongly governed by the limitation of forged steel vessels with regard to the maximum diameter for manufacture and transport. This limitation can be overcome by PCIV, which are composed of pre-fabricated segments compressed by axial and circumferential tendons. Leak-tightness is provided by an inner liner bolted to the cast iron structure. The segments can be easily transported to any place and have to be assembled and prestressed on site. The multi-tendon arrangement excludes any possibility for sudden burst, as already proven by experiments. Large cracks cannot progress as the compressive stresses are superimposed with the tensile stresses due to internal pressure.

Passive decay heat removal via the PCIV structure has also been tested and could be improved by integration of a passive vessel cooling system within the PCIV wall (see Figure 10). This could also improve the heat transfer mechanism and open the potential for enlarging the power size of a modular HTR or other types of modular reactors as well.

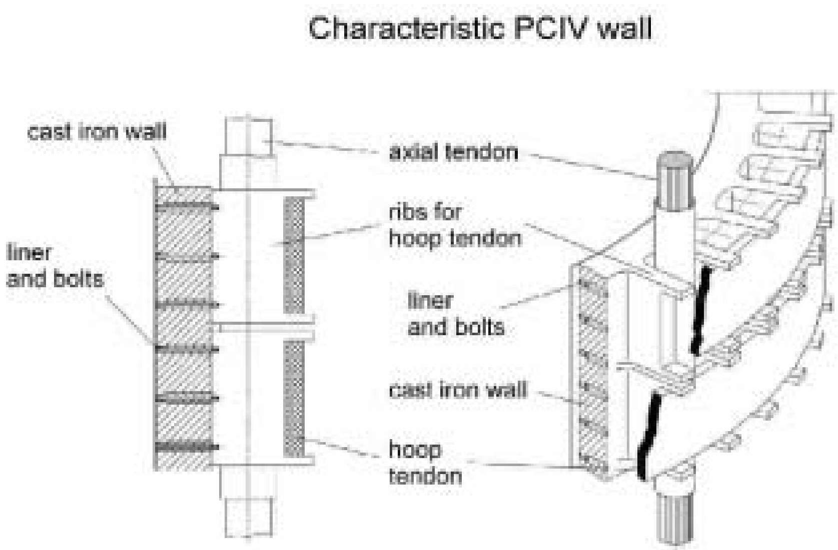


Figure 10. Wall structure of prestressed cast-iron vessel.

A test vessel with a 4 m inner diameter and 4.8 m height of the cavern is still available to investigate the behavior of the system. Another PCIV has been built and licensed for the THTR as a gas container for activation of the pneumatic absorber rod drives. The further development should target larger diameters and improved decay heat removal. R&D costs are about $7 million for 5 years.

Table 13. Long-term material development program R&D schedule and cost.

Year R&D	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	R&D Cost(k$)
Alloys																	25,000
Vessel																	5,000
Coating																	3,000
Graphite																	5,000
CFC																	5,000
Ceramics																	5,000
PCIV																	7,000
Total																	55,000

3.3.3.5 Passive Decay Heat Removal System. The passive reactor vessel cooling system for HTGRs, which surrounds the reactor pressure vessel (as shown in Figure 11), is designed to remove decay heat during a pressurized or depressurized accident. The Japanese HTTR has a vessel cooling system that is cooled by forced circulation of water; it is not, however, a completely passive system.

To design the vessel cooling system, it is essential to reliably predict the onset of hot spots on the components for the protection of such important components as biological shields, which are heated up by upward natural convection of hot air or by the penetration of high-temperature components during accident conditions. R&D using a scaled mock-up is proposed to demonstrate the cooling performance and integrity of the passive vessel cooling system that is cooled by natural circulation of air. It is also proposed to investigate the validity of the design and evaluation method, which can reliably predict the cooling performance of the system and the onset of hot spots on the components. The R&D schedule and costs are as follows:

Table 14. Development program R&D schedule and cost.

	1	2	3	4	5
Scaled mock-up demonstration tests	Design	Fabrication	Test		Evaluation
Numerical evaluation	Code-Code & Code-Experimental Benchmarks				
Budget (k$)	1000	14000	2000	2000	1000

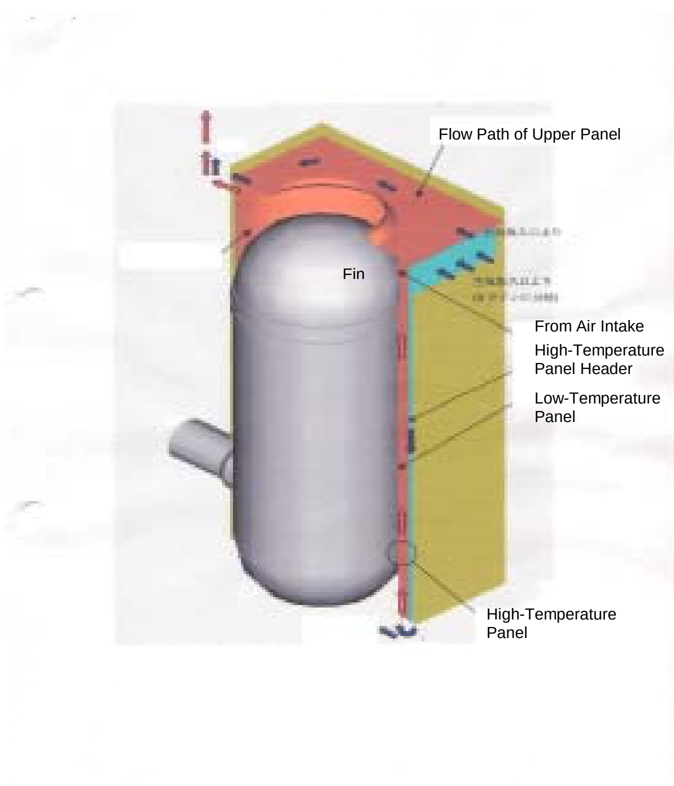


Figure 11. Drawing of passive vessel cooling system.

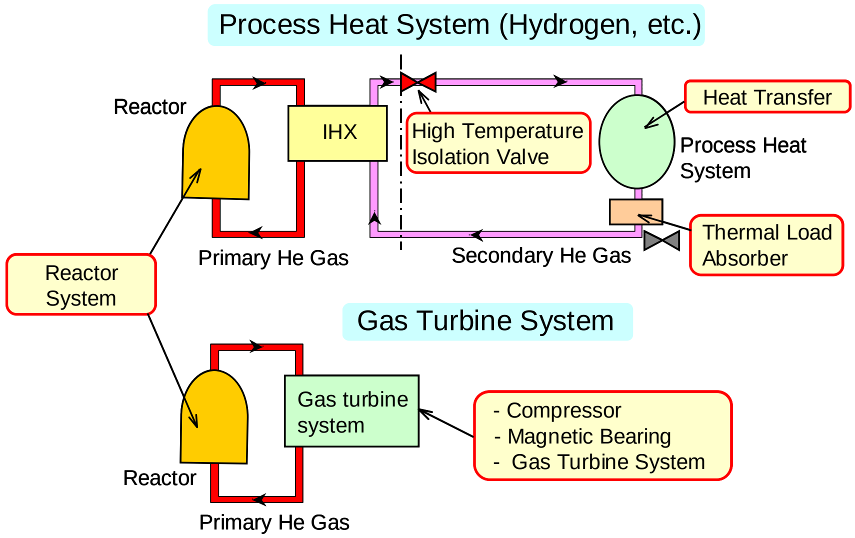


Figure 12. Conceptual separation of the reactor systems for nuclear heat application and for electricity generation.

3.3.4 Balance of Plant, Energy Products, and Process Heat Applications

3.3.4.1 Electricity Generation System. To improve maintainability, the potential option of locating the generator outside of the Gas Turbine Generator (GTG) vessel should be investigated. In this option, the maintainability and economical advantage will be highly improved because radioactive contamination of the generator can be avoided. It is important to develop an item to seal the penetration for the rotating shaft in the GTG vessel. Vertical orientation of the GTG is a potential option with potential economic advantages. Rotor dynamics stability, bearing technology, and maintainability should be investigated for this option. As shown in Figure 13, thermal efficiency increases with gas-turbine inlet temperature. As examples, the thermal efficiency is about 46 and 50% for 850 and 950°C at core exit coolant temperatures, respectively.

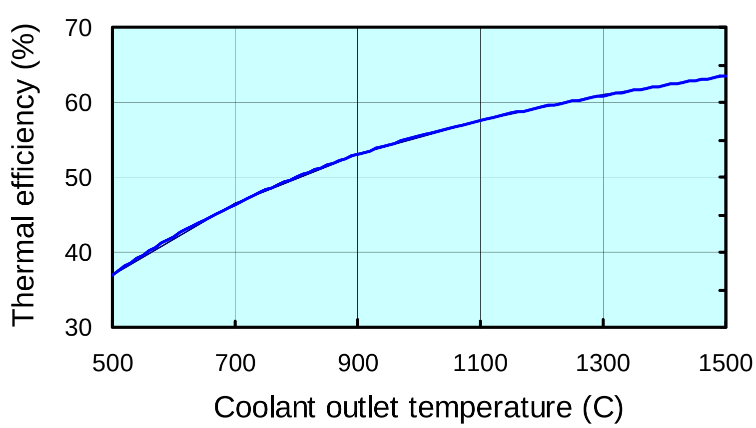


Figure 13. Thermal efficiency versus gas-turbine inlet temperature in electricity generation with VHTR.

3.3.4.1.1 *Compressor Model Performance Test*—R&D needs are as follows:

- A design method for high-efficiency compressor:
  - High hub-tip ratio → Energy loss by boundary layer separation, and tip clearance loss
  - Blade design to reduce losses

The following efforts are needed to address the above needs:

- Blade design study using 3-D CFD
- Sealed compressor model tests (see Figure 14)
- Test of aerodynamic performance and code verification
  - Surge characteristics
  - Thrust force by rotating blade
  - Inlet and outlet loss
  - Test model
- 4 stages, boss ratio: 0.88, 10800 rpm
  - Start-up characteristics verification

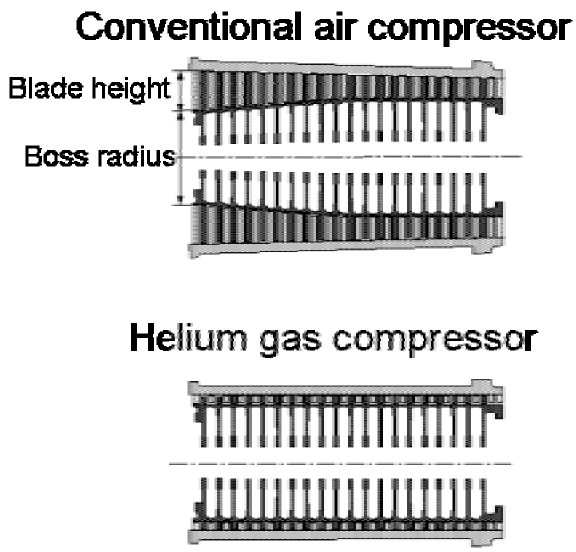


Figure 14. Comparison of compressors.

3.3.4.1.2 *Development of Magnetic Bearing*—R&D needs are as follows:

- A design method for turbo-rotor on magnetic bearings
  - Bending vibration of rotor
  - Damping control of magnetic bearing
  - Isolation of vibration by flexible coupling
- Catcher bearing
  - Structural integrity at touch down

– Rotor dynamics during coast down 1/3-scale Model Test for Verification

The following efforts are needed to address the above needs:

- 1/3-scale Model Test for verification
  - Magnetic bearing performance
  - Magnetic bearing controllability
    - 1st and 2nd bending vibration
    - Coupling of 2-span rotor
    - Validity of flexible coupling
  - Catcher bearing performance
- Establishment of full-scale rotor design

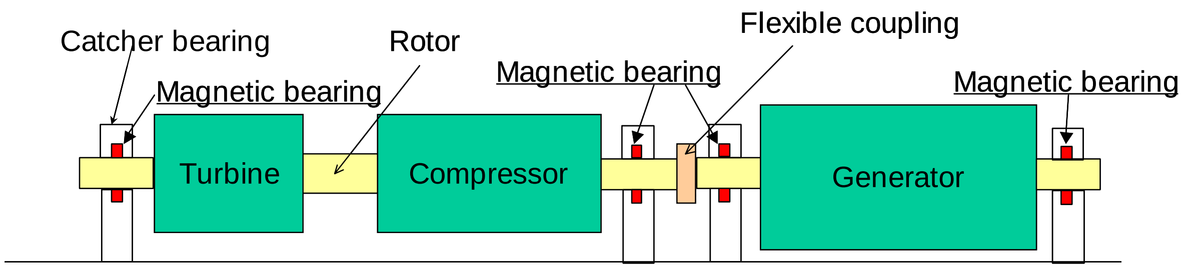


Figure 15. Full-scale rotor design for gas turbine system.

3.3.4.1.3 Turbine System Operation and Control Test—R&D needs are as follows:

- Operation and control methods for gas turbine power conversion system
- Acquisition of kinetics data on closed cycle gas turbine system.

The following efforts are needed to address the above needs by using helium gas loop:

- Construction of 1/5-scale integrated model turbine, compressor, generator, recuperator, precooler, intercooler, and electric heater in helium gas loop
- Development and verification of computer code
- Acquisition of data on system operational and control characteristics for
  - Normal operation
  - Startup and shutdown
  - Partial load operation
  - Transients
  - Turbine trip, etc.
- Verification of full-scale gas turbine.

3.3.4.2 Nuclear Heat Application. Both modular HTRs (GT-MHR and PBMR) have the potential to be coupled to several different power conversion systems for electricity production, cogeneration of electricity and heat, and to provide high-temperature process heat for a variety of nonelectric applications. They are taken here as synonym for both types of Modular HTR (MHR) irrespective of the type of core. One of the most interesting of these process heat applications is the production of hydrogen by thermochemical water-splitting (H2-VHTR). Hydrogen can be a significant market for nuclear energy. In the long term, the potential market for hydrogen is more than twice the market for electricity. Further, there is an immediate market for hydrogen in U.S. chemical processing industries that nuclear energy could help fill. Nuclear energy can provide a long-term, stable source of hydrogen at reasonable cost. Nuclear production of hydrogen will be the “enabling technology” for the Hydrogen Economy.

There are three principal methods for nuclear hydrogen generation (see Figure 16):

1. Water electrolysis and highly efficient electricity generation
2. Steam reforming of methane and light hydrocarbons
3. Thermochemical cycles.

Steam reforming of natural gas is the dominating industrial hydrogen production method and can also be coupled to an HTR to substitute the process heat for this endothermic process and gain much higher yields (see Figure 16). For evaluation of the different hydrogen production methods, it is necessary to recall elementary thermohydraulic rules, which govern the efficiency of thermochemical processes for hydrogen production as well as that of electricity generation in combination with successive electrolysis.

Both processes shown in Figure 17 are governed by the very similar Carnot law dependent only from the upper (T_h) and lower (T_c) operational temperature as well as from the dissociation temperature (T_d), which is 4309 K for autothermal water splitting.

$$\eta = [(T_h - T_c) / T_h] * [T_d / (T_d - T_c)] \ .$$

It can also be seen that the electricity needs are considerably reduced if the electrolysis is performed at higher temperature levels.

This formula has a direct impact on the choice of technical options:

- Operational temperatures of the process and the heat source should be as high as technically feasible
- Compared to high temperatures, thermochemical processes or electricity generation at lower temperatures will always be inferior with regard to the thermal efficiency η
- Inasmuch as the required dissociation temperature is too high, water splitting always needs several successive processes to provide the dissociation energy (i.e., electricity generation plus electrolysis [plus heat] or a follow-up of different endothermic and exothermal chemical processes at lower temperatures).

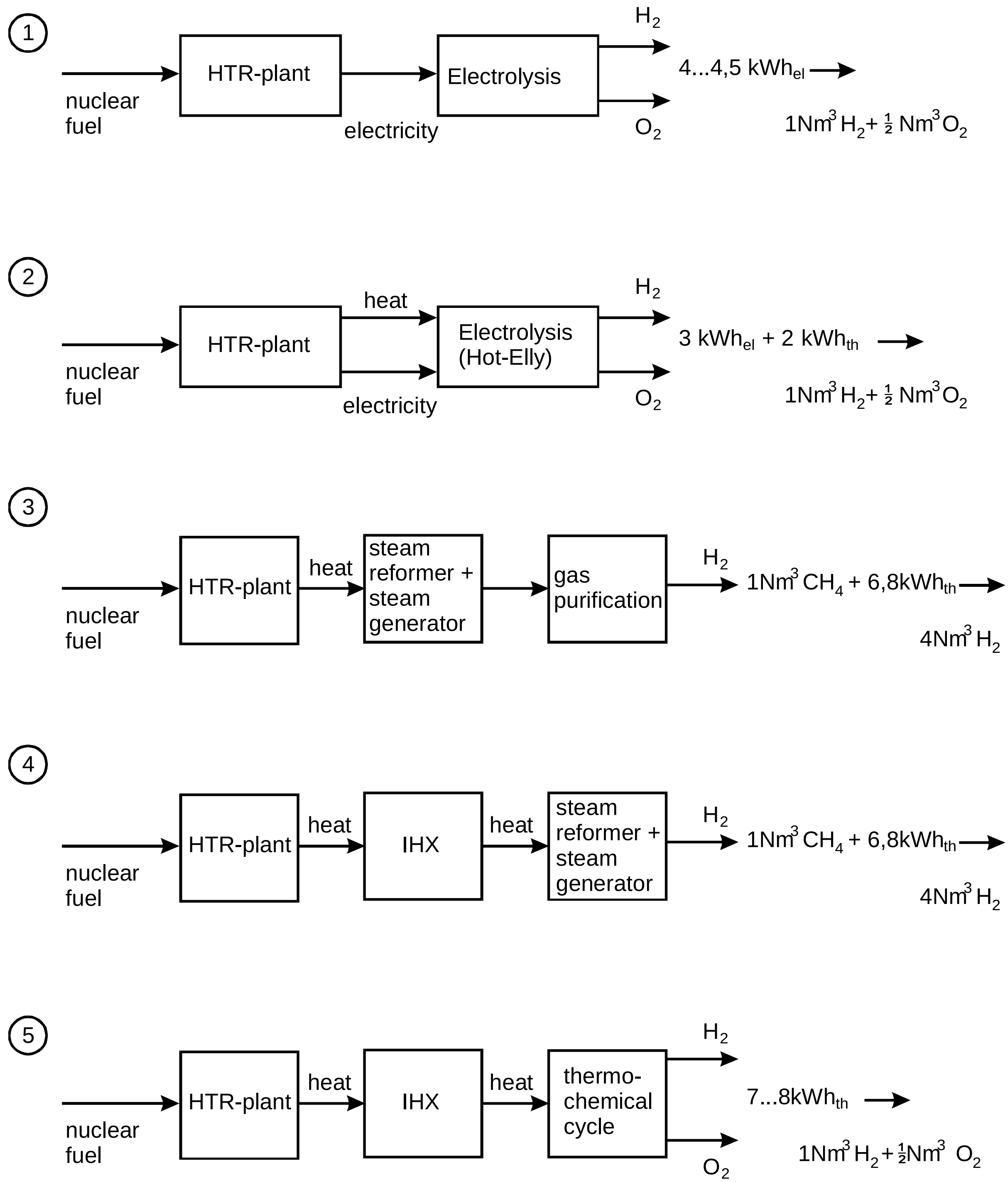
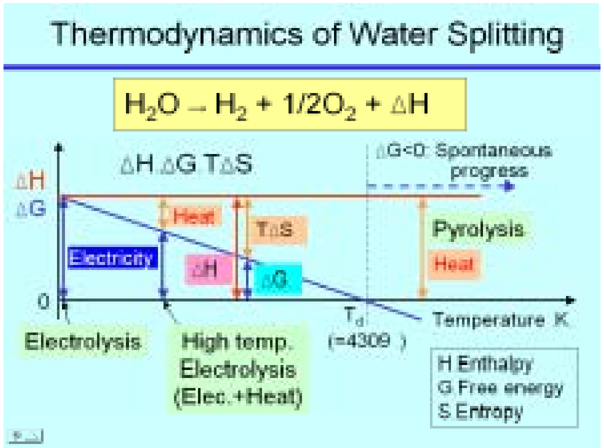
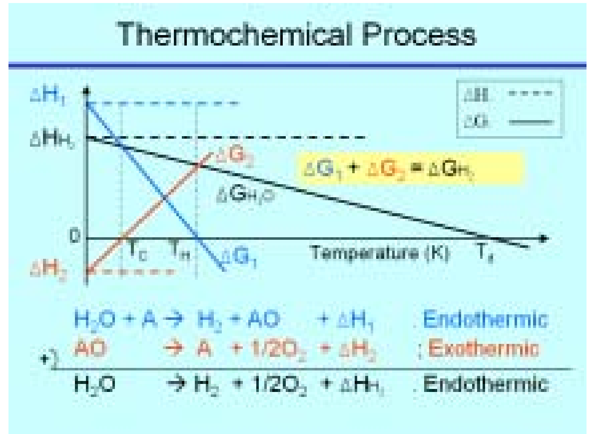


Figure 16. Energy input and yields of hydrogen production methods.



Case A: ELECTROLYSIS



Case B: THERMOCHEMICAL WATER SPLITTING

Figure 17. Comparison of electrolysis to thermochemical water splitting.

As a result, the final conversion efficiency is independent on any route of conversion technology on the assuming the same conditions for input and output.

Enhancing the efficiency for electricity generation and for electrolysis by increasing the temperature is one option for hydrogen production. Electrolysis can be done remotely and decentralised or with direct coupling to the reactor using high-temperature steam (HOT ELLY). This process can benefit from fuel cell development, which is the inverse process using the same functional elements.

Electricity (electrons) and hydrogen (protons) form complementary and synergetic options for transferring and storing energy for different end-uses. In a certain way, they are interchangeable, although conversion losses occur. Together, both offer much more flexibility in optimising energy structures (e.g., substitution of natural gas fired peaking plants by hydrogen). Hydrogen or hydrogen-rich liquid fuel (e.g., methanol) can be converted to electricity for transport purposes via fuel cells. Decentralised hydrogen production systems can be established via electrolysis if cheap (CO_2-free) electricity is available (e.g., off-peak nuclear power). These applications can be started using present nuclear power plants for off-peak hydrogen generation and, in the future, continued by highly efficient HTR with direct, indirect gas turbines, or combined cycle plants delivering the process steam at the necessary temperature/pressure level (e.g., for hot electrolysis or thermochemical water splitting, as well).

Another efficient and cost-effective way to produce hydrogen using nuclear energy is to use high-temperature heat provided by a reactor in a thermochemical water-splitting cycle, where a set of coupled chemical reactions decompose water into hydrogen and oxygen. Energy is input via endothermic, high-temperature reactions, and rejected via exothermic, low-temperature reactions. The chemical reagents are recycled within the process, and the net effect is to convert heat energy and water into hydrogen energy.

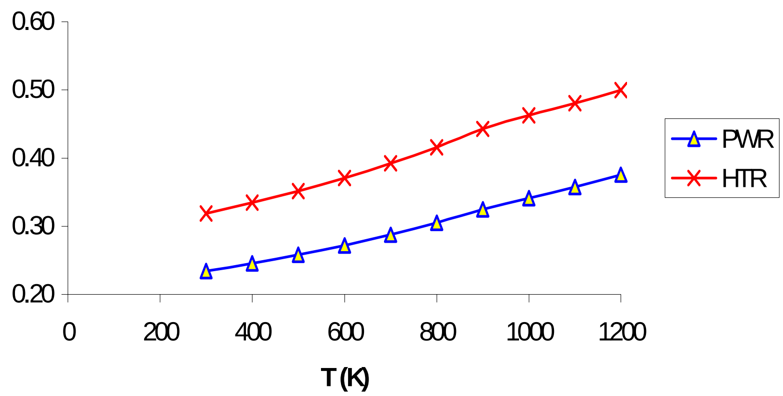


Figure 18. Efficiency of hot electrolysis using electricity from HTR or LWR /CEA-figure.

The VHTR is uniquely suited for coupling to the IS thermochemical water-splitting cycle for hydrogen production. The high-temperature heat makes possible the production of hydrogen at high efficiency (~50%) and reasonable cost (~$1.30/kg of H_2). The H2-VHTR will be coupled to the chemical process by an IHX loop. The helium primary reactor coolant will pass through an IHX and transfer heat to the intermediate loop coolant stream. This stream will transport the heat to the chemical plant, where it will be transferred through heat exchangers into the process working fluids. For best performance, the IS cycle needs to operate at a peak temperature of about 830ºC. To deliver heat at this temperature, the reactor outlet temperature needs to be raised about 100ºC from the 850ºC of the reference GT-MHR to 950ºC for the H2-VHTR. It is intended that, to the maximum extent possible, the NHSS for the H2-VHTR will be the same as for the GT-MHR or PBMR. The H2-VHTR will thus rely on and get significant benefit from the development activities described in the sections above for the GT-MHR and PBMR.

The H2-VHTR has additional development needs in three areas: (1) MHR incremental development, (2) intermediate heat transport loop, and (3) water-splitting process. In each of these areas, it is possible to build on significant previous effort. The development base and the additional development needs for these three areas are described below.

3.3.4.2.1 Interface technologies—The following interface technologies are needed between a reactor and heat application process:

1. Simulation test with a scaled facility of the HTGR hydrogen production system:
   a. *Status*—Simulation tests are necessary for licensing the HTGR hydrogen production system.
   b. *Time duration*—5 years.
   c. *Total R&D cost*—design and simulation tests, $20 million.
   d. *Technological difficulty*—None.
2. Demonstration testing of the HTGR hydrogen production system – A technique for connecting a hydrogen production system to an HTGR is required. Absorption of thermal disturbance of the heat application system to separate the kinetic characteristics of the heat application system from those of the reactor, operation procedure (especially the startup and shutdown phase), and so on should be investigated and developed.
   a. *Status*—Detailed design, licensing, construction of hydrogen production process and its connection to the reactor, and demonstration test will be performed.

b. *Time duration*—10 years.

c. *Total R&D cost*—detail design and licensing, construction, and demonstration test, $150 million, not including a nuclear reactor.

d. *Technological difficulty*—Some component tests require a demonstration or prototype. The use of HTTR for nuclear steam reforming is the most straightforward strategy to demonstrate nuclear hydrogen generation on an industrial scale within the next decade. The yield from natural gas will be enhanced by 30-40% by using a technique that is currently deployed in large scale. It can be assumed that this technique will be most economic as long as natural gas will be available at a reasonable price.

3.3.4.2.2 Thermochemical IS Process for Water Splitting—The concept of using thermochemical water splitting (a set of chemical reactions to separate water into hydrogen and oxygen at moderate temperatures) was developed in 1964 by James Funk, now at the University of Kentucky. Researchers at GA conceived several water-splitting cycles in the 1970s, the most attractive being the IS cycle. Iodine and sulfur-dioxide are added to water in an exothermic reaction that creates sulfuric acid and hydrogen iodide (1). These are immiscible and readily separated. The sulfuric acid can be decomposed at about 850°C, releasing the oxygen and recycling the sulfur-dioxide (2). The hydrogen iodide can be decomposed at about 450°C, releasing the hydrogen and recycling the iodine (3). The net effect is the splitting of water into hydrogen and oxygen (4).

$$2H_2O + SO_2 + I_2 \rightarrow 2HI + H_2SO_4 \qquad 100°C \qquad (1)$$

$$H_2SO_4 \rightarrow SO_2 + H_2O + {}^{1}/_{2}O_2 \qquad -900°C \qquad (2)$$

$$2HI \rightarrow I_2 + H_2 \qquad 450°C \qquad (3)$$

$$H_2O \rightarrow H_2 + O_2 \qquad (4)$$

The whole process takes in only water and high-temperature heat and releases only hydrogen, oxygen, and low-temperature heat. All reagents are recycled; there are no effluents. Each of the major chemical reactions of this process was demonstrated in the laboratory at GA. The IS cycle is projected to have an overall efficiency of about 50%. GA, Sandia National Laboratories (SNL)-Albuquerque and the University of Kentucky currently have a 3-year Nuclear Energy Research Initiative (NERI) grant to study nuclear production of hydrogen by thermochemical water-splitting. This team reviewed the available world water-splitting literature, identified and evaluated 115 different thermochemical cycles, and chose the IS cycle coupled to the GCR as the best suited for near-term application of nuclear energy to efficiently produce hydrogen. The GA-SNL-University of Kentucky team is now doing the detailed chemical flow sheet design of the IS water-splitting cycle matched to an MHR. It is estimated that the plant will achieve about 50% efficiency and may be able to produce hydrogen for as little as $1.30/kg of H_2.

The next required step in developing the nuclear-coupled IS process is to design, build, and operate a laboratory-scale, completely integrated, closed-loop IS experiment, driven by a nonnuclear heat source. This will take in water and simulated nuclear heat, and release hydrogen and oxygen at about 1-10 liter per hour. It will provide a convincing proof-of-principle that the nuclear-matched IS cycle is viable, and will allow verification that the chemical reactions indeed mesh together into a closed cycle and confirmation that reagent cross-over, impurity build-up, and hydrogen impurity levels are well understood and controlled. Some additional chemical data are needed to design the most efficient IS cycle. Thermodynamic equilibrium data are needed for the three-phase equilibrium of HI, I_2 and H_2O before a large-scale hydrogen plant should be built. These two essential steps will require about 4 years and $6–8 million to complete.

Following process development, a pilot plant will need to be constructed using fully prototypical materials and technologies. The pilot plant would operate on nonnuclear heat, simulating heat transfer from a nuclear reactor. It would use only a fraction of the 600 MWt of heat of a typical MHR module to produce ~100–10,000 m^3/hr of H_2. This pilot plant would demonstrate the technologies and materials of a full-sized plant, verify plant control systems and operability, and confirm materials performance. To design, build, and operate a 10-30 MWt pilot plant will take 4-5 years and approximately $65 million.

Table 15. IS thermochemical water splitting process R&D schedule and cost.

Activity	1	2	3	4	5	6	7	8	9	10	Cost ($K)
Laboratory Loop Design, Construction and Operation											6,000
Chemical Data Measurements											500
Pilot Plant Design, Construction and Operation											65,000
Total											71,500

The R&D mentioned above is complementary to the ongoing work in Japan on the UT-3 and on the IS processes.

- *Status*—The R&D items for the IS process are as follows:
  - Continuous hydrogen production technique
    - Laboratory scale study of continuous hydrogen production
    - Related chemical data
    - Bench-scale and pilot-scale experiments, then demonstration test with nuclear heat.
  - Separation technique to improve the HI decomposition step for high efficiency
- Study on application of membrane technologies.
  - Selection of materials for constructing a large-scale plant, which can be used in the corrosive process environments (see Figure 19)
    - Corrosion tests of commercially available materials
    - Effect on mechanical properties
    - Surface modification.
- *Time duration*—10 years.
- *Total R&D cost*—
  - Basic study and design of facility for bench-scale test: $7 million
  - Bench-scale test: $30 million.
- *Technological difficulty*—The technologies for attaining thermal efficiency higher than electrolysis are in the basic stage.

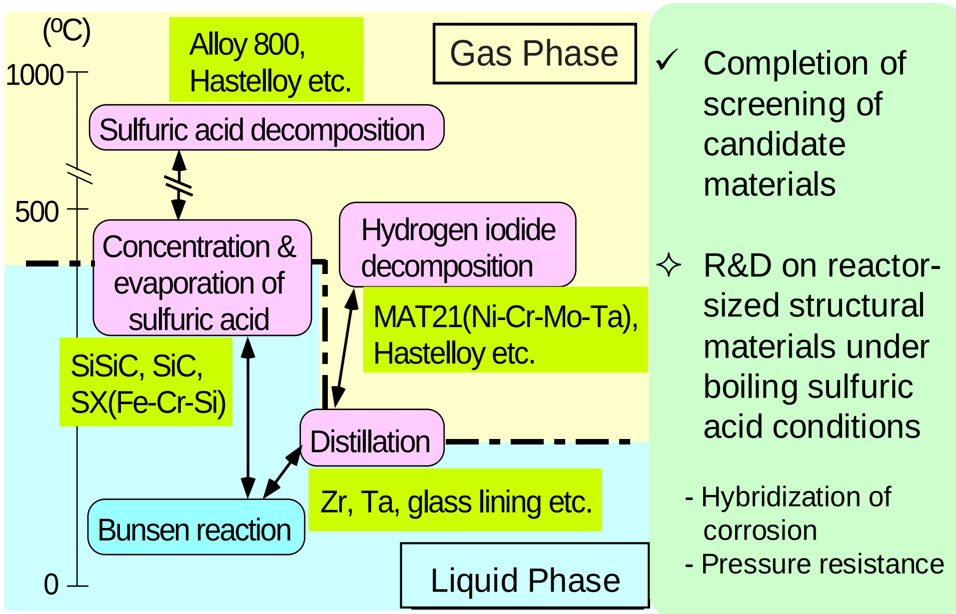


Figure 19. Process for selecting materials that can be used in large-scale plants.

***3.3.4.2.3 Steam Reforming of Methane Using Nuclear Heat*—**The endothermal steam reforming reaction of methane takes place in a wide temperature range of 500 to 800°C and above in the presence of a nickel catalyst:

$CH_4 + H_2O \rightarrow$ $CO + 3H_2$ - 206 KJ/mol

$CH_4 + 2\ H_2O \rightarrow$ $CO_2 + 4H_2$ - 165 KJ/mol

High temperatures, low pressures, and low H_2O/CH_4 ratios are suited to get low residual CH_4 contents in the reformer gas. CO which is still contained in the reformer gas, is converted by the exothermal shift reaction:

$CO + H_2O \rightarrow$ $CO_2 + H_2$, +41 KJ/mol

Figure 20 includes a principle flowsheet in which all the heat for the reforming process, for the steam production, for gas purification, and for gas compression can be gained from the helium circuit of an HTR.

The steam reformer uses the temperature of the helium (40 bar) between 950 and 700°C, and the steam generator uses that part of heat between 700 and 250°C. The feed gas ($H_2O/CH_4 \approx 3/1$, $p \approx 40$ bar) is preheated until about 500°C and reformed with a maximal process temperature of 800°C. In this first step, 85% of the methane is then converted. Using reformer gas heat for preheating the feed gas, shift conversion, and methanation are the next steps to producing hydrogen. CH_4, as raw material, is converted 100% to hydrogen; the total efficiency, including the nuclear heat, is around 65%. The steam reformer and the steam generator can also be arranged in the IHX circuit, as is the case in the Japanese HTTR project.

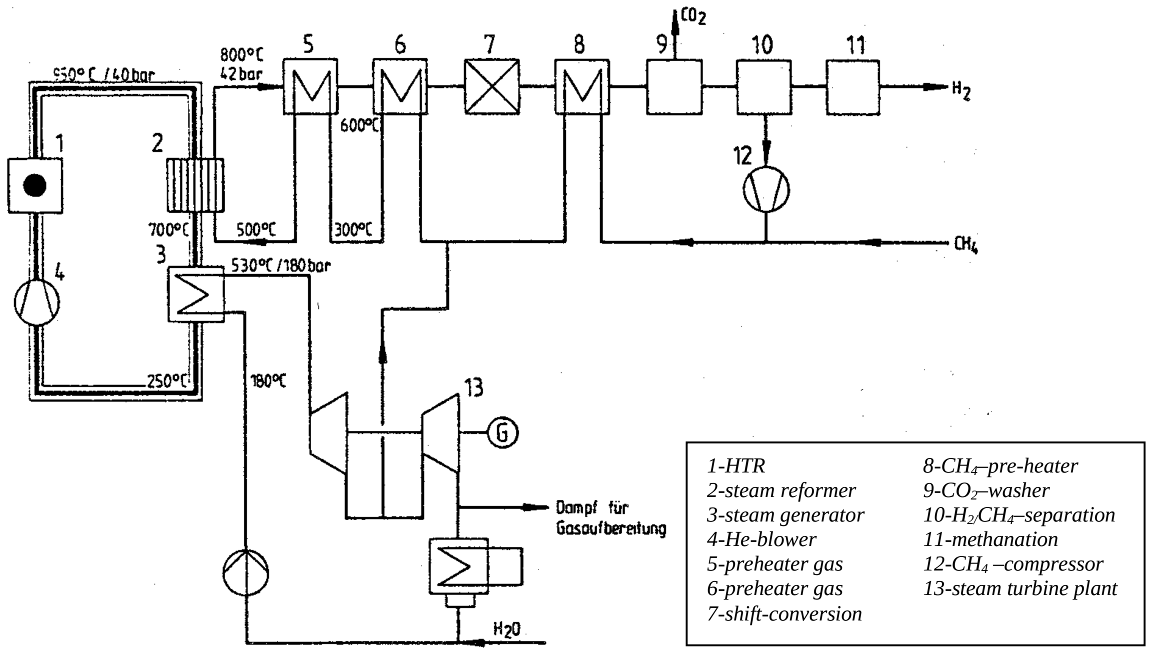


Figure 20. Principal flow sheet for steam reforming of methane using nuclear heat.

3.3.4.2.4 R&D Issues for VHTR Deployment in Refineries, Petrochemistry, and Aluminum Production—Large amounts of process heat are consumed in crude oil refineries, in petrochemistry, and in aluminum production. These applications have been identified as most promising for VHTR within the next decades, and make use of the basic developments for IHX, valves, hot gas ducts etc.

Table 16 shows the thermal power demand in a 6 million-t/y crude oil refinery. It can be seen that most of the heat is consumed at a temperature below 540°C, which can be delivered by high-quality steam. The hydrogen demands are comparably small but will steadily increase depending on the crude oil grades. A VHTR with IHX steam reformers and combined cycle could easily provide all necessary process energy for large refinery complexes.

Table 16. Thermal power demand of a refinery (6 million t/y).

Process	Heat (MW)	Temperature	Pressure (bar)
Crude oil distillation	117	230-370	1
Vacuum distillation	46	230-385	30 torr
Propane de-asphalting	53	50-80	35
Vacuum residue distillation	20	340-385	100
Vacuum gas oil desulphurisation	17	340-385	40
Middle dist. desulphurisation	15	340-385	25
Gasoline desulphurisation	12	340-385	30
Gasoline reformer	53	430-540	30
Hydrogen generation	5	820-850	30
Effluent water cleaning	3	20-60	1
Steam generation	112	20-500	20
Total power demand	**453**		

It is proposed to make a detailed design for a refinery heated by VHTR to increase the yields and decrease the CO_2 emissions related to the refining process. The work should be completed in 3 years at a cost of $0.5 million per year.

Refineries are often combined with petrochemical plants for naphtha production. The total installed world naphtha processing capacity is about 92.7 million t/y. The heat and electricity required to process 1.5 million t/y Naphtha is about 234 MWth. The maximum temperature to crack naphtha into ethylene, propylene, etc., is approximately 840°C. By use of an IHX, the VHTR operational temperatures should be around 950°C.

The necessary development should address:

- Primary He—Secondary He Heat Exchanger (950°C–900°C, 300°C–250°C)
- Secondary He—Steam Cracker (900°C–800°C, 840°C–600°C)
- Secondary He—Super Heater (800°C–290°C, 600°C–250°C)
- Secondary He—Naphtha Evaporator (290°C–200°C, 250°C–120°C)

The heating of the naphtha with helium from 600 to 840°C in a fraction of a second has been theoretically checked and appears to be feasible.

System optimization for a combined energy supply system of a petrochemical plant and a refinery should be done together with process engineering companies and potential users; development of specific components, as mentioned above, would follow. System optimization should take 5 years at a cost of $0.5 million per year, with development of specific components lasting five additional years at a cost of $2 million per year.

Another important industrial production that can be coupled to VHTR is aluminum oxide production. Currently, bauxite (being the raw material) is being transported over large distances to places that offer cheap process heat. Afterward, the aluminum oxide is transported again to the sites that offer cheap electricity to produce metallic aluminum by electrolysis. The total installed world production capacity (as of 2001) is 56,326,000 t/y. In 1990, it was only about 40 million t/y, which is an increase of approximately 40% in the last decade. Heat and electricity required for 300,000 t/y is ~ 400 MWth. The number of plants with production capacity over 300,000 t/y is about 55. The maximum process temperature can be kept at 950°C.

Heat exchanger development should address:

- Primary He—Secondary He Heat Exchanger (only Primary Component)
- Secondary He (900–680°)—Fluidized Powder Heat Exchanger (850–190°)
- Secondary He (680–419°)—Steam Generator (500–170°)
- Secondary He (419–250°)—Liquid Salt Heat Exchanger (400–240°)

This application should also be taken into consideration by optimizing the adaptation of VHTR and aluminum oxide production plant and developing the specific components. Plant optimization will take 5 years at a cost of $0.5 million per year, with development of specific components lasting 5 years at a cost of $2 million per year.

These examples show that there are already near- to medium-term applications beyond dedicated electricity generation that could be combined with VHTR. Specific components have to be developed for each process individually but can profit from existing process engineering experience. The NHSS, the IHX, and the isolation valves could be largely standardized. These applications require only small- to medium-sized VHTR (200-300 MW per module) with high reliability. Substitution of the process heat by VHTR will enhance the yields and reduce the CO_2 emissions. In addition, it must be remembered that oil sands and oil shale that might be used in the intermediate future will consume much more heat for retorting and refining than is the case with crude oil.

3.3.4.3 Components.

3.3.4.3.1 Modular High-Temperature Reactor Incremental Development—The need for a reactor outlet temperature that is ~100ºC higher and the low pressure of the thermochemical processes will necessitate some design changes, especially for the block core design due to the higher fuel temperatures in the fuel compacts. The basic approach will be to raise the core outlet temperature without exceeding peak fuel temperature limits (1230ºC). A few small design changes will make this possible without raising the critical temperature limits on the fuel or the reactor structural materials. These design changes may include use of a different core inlet temperature, use of smaller diameter fuel rods, more thermal insulation in the cross duct, or other changes that would not change the basic reactor technology and may not require any substantive R&D efforts.

A design activity is proposed as part of the NERI activity to do a serious conceptual design for the H2-VHTR. This design effort will evaluate and select the design changes from the base GT-MHR that will be needed for the H2-VHTR. This includes the core mechanical and thermal design, the pressure vessel and cross duct design, the circulator design and a safety evaluation of the finished H2-VHTR. This design work will allow reevaluation of the current assessment that relatively little specific R&D beyond the circulator development will be needed for the MHR portion of the H2-VHTR.

- *Circulator Development*—The helium circulator for the primary loop of the H2-MHR will be located in the IHX cavity and will be virtually identical to the circulator designed for the steam cycle MHTGR. It will also parallel the development effort for the SCS circulator in the generic portion of the MHR program. This includes impeller aerodynamic and acoustic tests, and prototype tests of the circulator unit in a HPTF under reactor conditions to verify the design. This development effort can be accomplished within three years at a cost of $5 million, including test facility costs. The construction of the HPTF for the SCS circulator will be fashioned to accommodate H2-VHTR circulator development. Most of these adaptations can also be used for the PBMR-based designs.

3.3.4.3.2 Intermediate Heat Transport Loop—The high-temperature heat from the VHTR will be coupled to the IS thermochemical water-splitting cycle by means of an intermediate heat transport loop. The working fluid of the reference concept will be pressurized helium or nitrogen at a pressure intermediate to the 70 atm of the MHR and the 8-20 atm pressures of the IS process. In former NPH projects, the primary reactor pressure was reduced to about 40 atm to limit the differential pressures. Reactor primary loop helium will enter the IHX, located in a well adjacent to the reactor, at 950ºC and will return to the reactor at about 500ºC. The intermediate loop will operate at an upper temperature of about 900ºC and a lower temperature of about 350ºC. Lower helium return temperatures could ease the design and material choice for the pressure vessel. The intermediate loop piping will come out of the IHX well and exit the reactor building to transport the heat to the hydrogen process plant. Heat will be transferred into the water-splitting process via heat exchangers that are part of the hydrogen production process. The intermediate loop circulator will be located out of the IHX well and will be very similar to that used in the primary helium loop. The development effort described in the above section for the primary loop helium circulator will provide all the information needed for the intermediate loop circulator as well.

3.3.4.3.3 Intermediate Heat Exchange—The IHX between the primary and intermediate helium loops will be a fairly conventional design. In the former German PNP project, helix and U-tube IHX have been developed and tested in a 10 MW-scale. The IHX of the HTTR has the same power size. Currently, the "printed circuit heat exchanger" configuration similar to those manufactured by Heatric Division of Megglitt PLC of the U.K. are favored because of their compact size and high efficiency. These heat exchangers are not yet commercially available at the temperature range needed for the H2-VHTR. Additional design work will be needed, and it is possible that an ASME code case will have to be developed for the materials needed for hydrogen production. It is anticipated that conventional high-temperature metal alloys or, alternatively, code-qualified nickel-chrome Alloy 617 will meet the requirements for nonnuclear use at up to 980ºC. Assuming the data are available from nonnuclear qualification, this would be a 5-year $500,000 effort and a 1:1 scale testing of critical components for approximately $5–10 million. When the heat is applied to hydrogen production, nuclear heat from the core reactor should be transferred into the secondary helium through the IHX for safety reasons. The IHX needs a highly reliable boundary between the primary and the secondary helium coolants as well as high thermal efficiency and compactness. For these requirements on the IHX, plate-fin type compact heat exchangers should be developed.

- *Objective*
  - Heat transfer augmentation
  - Development of compact steam reformer, heat exchanger, etc.
- *Present Status*
  - Basic characteristics at low temperature were obtained with fine-mesh in flow conduit.
- *Research Schedule*
  - Basic tests for 3 years, $300,000
  - Design for 5 years, $20 million.

3.3.4.3.4 High-Temperature Isolation Valve—Development of the isolation valve is a key issue in the hot and cold gas ducts of the intermediate circuit. Their objectives are to separate the reactor system with nuclear standards from a process heat system with nonnuclear standards (see Figure 12). Testing was completed on coating material of the valve sheath, focusing on antiseizure and adhesion performance for use in HTTR. Integrity test of the valves in full scale will be necessary. Results from former German tests of two different types of valves and of diverse hot gas ducts should be retrieved. The integrity tests against repetitive operation of the high-temperature isolation valve have to show the reliable performance after multiple activations:

- Mock-up model test for 4 years, with $3 million
- Demonstration test with HTGR for 3 years, with $10 million.

Material development for valves is rather tough, but if no material is found for the sheath, another way, such as changing the valve for a certain period, could be taken into account.

3.3.4.3.5 Thermal Load Absorber—Nonnuclear heat processes would generate thermal disturbances resulting from disturbances of flow rate and pressure. These disturbances might change the inlet temperature at the IHX in the secondary helium gas loop. Temperature change greater than the limitation value would lead to shutdown of the HTGR. Thus, thermal disturbances that are not large but may stop the HTGR should be absorbed with a thermal load absorber.

- *Objective*—Develop a thermal load absorber with high-temperature latent heat storage technology.
- *Present Status—*
  - Increase effective thermal conductivity of Phase Change Material (PCM) by absorbing PCM into porous materials.
  - Estimate reduction characteristics of thermal loads with PCM.
  - Conduct material tests to evaluate corrosion resistance of metals to PCM.
- *Research schedule—*
  - Material Test—3 years, $300,000
  - PCM Selection—2 years, $100,000
  - Heat Exchange Test—3 years, $500,000
  - Mock-up Model Test—3 years, $1 million.

3.3.5 Safety Concepts and Performance

The use of process heat from nuclear reactors in chemical processes requires short distances between the nuclear and the chemical plant. Compared with electricity and steam generating nuclear power plants, an additional risk is caused by this close proximity. After the accidental release of process gases, flammable clouds can be formed, which in the worst case can explode. However, in the licensing procedure it has to be proven that a possible explosion will not damage the containment building or other safety-related parts of the nuclear plant. Below-grade siting might help to reinforce the protection of the nuclear installations.

On the other hand, no major risk should be generated by the NHSS itself with regard to the highly populated industrial environment and high investments for the production plants. Any contamination of the conventional part or of the product has to be avoided. Thus, it can be concluded that an NPH plant has to fulfil much higher safety requirements than dedicated electricity generating nuclear power plants. Convincing safety demonstration and reliable exclusion of product contamination will be a prerequisite for entering into the NPH and hydrogen market.

Many refineries and petrochemical plants remain fully operational for very long periods, requiring a very reliable and redundant power supply. The optimization of NHSS designs for NPH might not tend toward increased power size, but rather toward simplicity, robustness, reduced shut-down periods, extended operational periods (e.g., on-line refuelling), large safety margins, reduced nuclear transports, ease of decommissioning, etc. As necessary power sizes will be much smaller than is the case in the electricity market. The design of VHTR should make use of this fact by transferring the margin in power size into the afore-mentioned objectives and seeking a high degree of standardization and modularization for the NHSS. The necessity of containment or inherent filtered releases, also in case of large breaks, will have to be re-discussed under the specific site conditions.

Reliable IHX and isolation valves have to safely separate the nuclear island from the conventional production in case of maintenance, repair, and incidents. In summary, the design has to respect the following requirements:

1. *Severe Accident Free Design*—Severe accident is prevented in the worst event. Radioactive releases have to be restricted to the nuclear island. Innovative filters, even to cope with large breaks, have to be developed if containments are to be avoided from cost reasons.
2. *Demonstrable Safety*—Safety is demonstrated in the first-of-a-kind reactor.
    a. Loss-of-coolant-flow accident simulation
    b. Control-rod-withdrawal accident simulation
    c. HTTR and HTR-10 have to be used for safety tests and code validation.
3. *Role of Probabilistic Risk Assessment*—
    a. To select the event categorized in Anticipated Operational Occurrences (AOO), DBAs, and BDBAs
        – Pipe break accident/DBA
        – Guillotine pipe break/BDBA.
    b. Doses evaluated in AOO, DBA and BDBA are nearly the same.
4. *Simplified Safety System*—
    a. Normal Operation
        – Hydrogen/Tritium transportation between the reactor primary coolant and the product hydrogen gas via permeation through the heat exchanger tube walls.
    b. Anticipated Operational Occurrence
        – Thermal transient from the hydrogen production plant to the reactor core due to malfunction of the hydrogen production process.
    c. Accident
        – Fire and explosion due to the accidental release of combustible materials, such as product hydrogen gas (and feed natural gas).
    d. Issues
        – Detonation condition for outdoor vapor explosion
        – Blast overpressure
        – Effect of radiation and blast on the integrity of the components
        – Missiles due to explosion
        – Code validation through benchmarking to other computer codes and experimental data
5. *Site-Related Safety Aspects*—A safety analysis should be made for a standardized design of the modular VHTR as well as for typical industrial sites (refineries, chemical production complexes, etc.). Minimal safety distances have to be determined with regard to:
    a. Loads on confinement structures from gas or vessel explosions, fires

b. Physical protection of the nuclear island (e.g., terrorist attack)

c. Determination of operational and accidental releases.

The safety distances have a direct feedback on the thermal hydraulic requirements of the design of hot gas ducts with minimal temperature loses between the heat source and the heat utilization system.

1. *Product Contamination*—If HTR-plants are applied for chemical processes, like hydrogen production, only a very low product contamination by tritium is tolerated (e.g., <10 pCi/g). The tritium could permeate from the primary circuit through the walls of the steam reformer or IHX to the secondary side. In the reactor, tritium can be produced by ternary fission (U-235, [n, f] T), lithium impurities in graphite components (Li-6 [n, α] T), B_4C in control rods (B-10 [n, 2] T), or from the He-3 fraction in the coolant (He-3 [n, p] T). The permeation rates of tritium through the walls of high-temperature heat exchangers have been measured dependent on temperatures, type of materials and the process conditions (steam reformers, steam generators, IHX). Selective filter systems to take tritium from helium circuits could also be developed. Oxide layers on the heat exchanger surfaces were found to significantly reduce the tritium transport through the walls. In situ oxide layers have shown a large inhibition of permeation in the temperature range of interest (T > 600°C).

The potential R&D on countermeasures starts from high-purity graphite with low Li, content to suppress the generation of tritium via improved coatings and purification methods, to selective filters.

Table 17. Safety concepts and performance R&D schedule and cost.

Year R&D	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	R&D Cost (k$)
1																	5,000
2																	5,000
3																	5,000
4																	5,000
5																	5,000
6																	5,000
Total																	25,000

3.3.6 Economics & Markets

Since competitive economics is attained in PBR and PMR, VHTR could reach stronger economics due to higher efficiencies. However, it should not be forgotten that the competing conventional systems have a much higher innovation speed than nuclear technologies. Cost escalations for fossil fuel and/or CO_2 taxes may improve the future economic situation for NPH. Nevertheless, the NPH designs should be capable of competing without relying on exhaustion of fossil energy resources and the related cost increases. This will require a very straightforward modularization and standardization of the NHSS and of heat utilization systems. Conventional standards should be used as far as possible. Combined cycle plants with gas and steam turbines coupled via IHX might be indispensable for achieving higher efficiencies and for offering the necessary steam qualities. Cost evaluations show that nuclear steam reforming is close to competitiveness for hydrogen production. Thermochemical cycles will have to compete with advanced high-temperature electrolysis. Both are more expensive than methane steam reforming as long as cheap natural gas is still available, since hydrogen is produced with methane burning heat of about 75% and nuclear heat of about 25% in the methane steam reforming process versus nuclear heat of 100% in the water splitting method. Higher prices for fossil fuel will lead to the use of ‘dirty fuel,’ like oil shale and oil sands, needing much more process heat, which can be provided by VHTR.

Cogeneration of electricity and heat or co-production of hydrogen, electricity and heat will improve the economics and lead to better efficiencies. The design is than determined primarily by the heat consumption. Detailed economic and market studies are necessary to assess the multi-product approach.

Realistic economic studies on diverse applications and energy cost scenarios must accompany the R&D to identify the targets for industrial deployment of nuclear energy in the nonelectric market being much larger than dedicated electricity generation. About $6.5 million should be spent for a few years to evaluate the inclusion of process engineering companies and potential users.

3.3.7 Major Codes

Required analytical codes for design and safety analysis, such as dynamics of reactor system (reactor and utilization process of gas turbine or hydrogen production process), neutronics, FP release, flammable gas explosion (including pool burning of spilled liquefied natural gas), dispersion of flammable gas, deflagration/detonation, pressure wave/shock wave propagation, air/water ingress, and so on, will be completed for PBR and PMR. Analysis of the influence of an explosion in open air to the reactor is especially required for safety analysis codes for hydrogen production. Data on fuel and materials obtained for VHTR must be set in the analytical codes. Modeling and some analysis code are very important to predict characteristics of fuel manufacture and irradiated fuel. Verification of these analytical codes is also needed. This effort will cost $3.0M.

The source term estimations for NPH applications must be much more reliable as no remote site will be possible. It can be expected that improved coated particle fuel will not contribute to the radiological releases even in case of a core heat-up accident. The deposition of radioactivity over long operational time and its potential remobilization will be more decisive. The modeling of the fission product behavior in the core and the validation of these models have to be improved because there are still rather contradictory findings.

3.3.8 Integration

3.3.8.1 ***Probabilistic Risk Assessment.*** Relating to safety, PRA is needed to optimize the VHTR design and to show low risk to the public and an exclusion of product contamination, which would endanger the acceptance of NPH. Cost for this effort is $3.0 million.

3.3.8.2 ***Demonstration Plant.*** A demonstration plant of the VHTR has to be constructed by making best use of HTTR and HTR-10, which should be performed as international projects. Early inclusion of chemical process engineering companies and potential users of NPH is essential for the success and focus of the R&D programs. This effort will cost $400 million.

3.4 Gas-Cooled Fast Reactor System

The GFR will meet the Generation IV objectives of sustainability (through use of a fast spectrum and full recycle of actinides), safety, economics (through simplified design leveraged on PMR designs and operation at high temperature), and nonproliferation (through use of proliferation-resistant fuel cycle technologies and possible integration of the fuel cycle on the reactor site).

The main characteristics of the GFR are high operating temperature, fast neutron spectrum, robust refractory fuel, direct conversion with a gas turbine, and integrated onsite fuel cycle. Its development approach is to rely, as much as possible, on technologies already used for the PMR concept, but with significant extrapolations to reach the objectives stated above. Thus, the concept calls for specific R&D beyond current and foreseen work on the thermal PMR, especially in the field of fuel and fuel cycle processes. A consistent, specific 10-year R&D program has been devised on the GFR feasibility assessment and conceptual design. The objective of the 10-year R&D program is to identify and resolve key feasibility issues. The R&D plan includes fuel and fuel cycle considerations, including reprocessing, reactor system structure materials, and safety. These elements are strongly linked. The R&D plan components all follow the same schedule with preliminary choices in 2002, screening from 2002 to 2005, confirmation of choices and optimization from 2006 to 2012, and design and selection of technologies (materials, treatment processes, fabrication, etc.) by 2012 to enter a demonstration phase. In parallel, calculation tools have to be developed and validated by 2007–2008 to perform detailed design studies of a prototype version of GFR envisioned as an international project to be put into operation before 2020.

3.4.1 Fuel Development

Current PMR designs rely on the use of specific, coated particle fuels encased in large hexagonal graphite blocks that provide for neutron moderation, heat transfer, and thermal inertia. These technologies are not adapted to the fast neutron spectrum of the GFR. Innovations are required to adapt the concept of fuel particle and the core layout (without large graphite blocks) to plutonium and other transuranic-bearing fuels, to allow operation with a fast neutron spectrum and to resist high levels of fast neutron fluence. Technology breakthroughs are needed to develop innovative fuel forms that preserve the most desirable properties of standard gas reactor fuel particles, including withstanding temperatures up to 1600°C with an excellent confinement of the fission products while accommodating an increased heavy nuclei content and withstanding fast neutron irradiation. Fuel development efforts must be conducted in concert with reactor design efforts so that a fuel that meets core design requirements and a core that operates within fuel limits is developed. Alternative coatings, buffer materials, or matrix materials will be identified and investigated. Innovative fuels, such as refractory fuels (CERCER) maintaining a strong confinement for fission products during normal and off-normal plant conditions, should also be evaluated. A reference fuel concept and two alternative fuel solutions will be considered. The development of the GFR fuel will require a gas-cooled experimental reactor with fast neutron capability for a validation in fully representative operating conditions. Key dates are:

- 2002 Select reference fuel and alternative solutions
- 2002–2005 Accumulate basic data concerning inert materials (coating, buffer, matrix) and actinide compounds (carbide, nitride, etc.) to be considered
- 2003–2004 Launch first irradiation experiments in fast flux (Phénix, Joyo) of inert materials and uranium fuels
- 2007–2012 Irradiate uranium, plutonium, and minor actinides, and optimizate fuel concept
- 2003–2005 Fabricate fuel subassembly mock-ups
- 2006 Confirm reference fuel option and design of subassemblies
- 2006–2010 Aero-dynamic characterization of subassemblies
- 2012–2015 Fabricate prototype core.

The capability for spent fuel treatment after short cooling times using either current, updated, or new processes will also be investigated. Fuel should, therefore, also be compatible with remote, simple, and compact technologies for actinide spent fuel treatment and re-fabrication before recycling.

3.4.1.1 Preliminary Assessment of Fuel Design by Analysis. Fuel forms and configurations to be considered and developed in this plan are governed by neutronic and thermal hydraulic factors, which are determined by the goal of high sustainability for the GFR concept and by other Generation IV criteria such as economics, safety, and nonproliferation. To achieve fast neutron spectra needed for high sustainability, a high heavy atom fuel density is needed; low absorption-low moderation material should be used in the core, and the fuel burnup should be at least 5% FIMA. There should also be an ability to incorporate the minor actinides into the fuel. The safety case will require a high level of fission product confinement, especially if the direct cycle will be utilized to increase the efficiency of the plant. Safety consideration will also dictate fuel forms and configuration that allow for efficient cooling with high conductive and convective heat transfer coefficients, both under normal and accident conditions, and a low flow resistance in the core. Resistance to corrosion by coolant impurities over the core lifetime would add to the safety and economic viability. Furthermore, nonproliferation thresholds would limit the fissile content to the range of 15-20%. Nonproliferation concerns also emphasize the approach of integrated onsite fuel cycling, requiring fuel forms compatible with an economical recycle process.

There are several potential paths for fuel development, involving both extensions of current technology and development of entirely new technology. Fuels under consideration are traditional coated particle fuels, intermediate designs of coated particles in an inert matrix, standard pin-type fuel with high-temperature cladding, and CERCER composite fuels (CERMET as back-up). Current gas reactor particle fuel technology is focused on the BISO and TRISO designs of the PMR and other high-temperature thermal GCRs. These design concepts will have to be extrapolated to fast fluence tolerant designs, which use dense carbide or nitride actinide compounds at a much higher loading fraction. In extrapolating from thermal reactor fuel designs to fast reactor fuel forms, the management of fission gas release, and the material fluence damage limits have to be evaluated in the selection of the optimum kernel sizes and shapes as well as which particular actinide compound is to be utilized. Composite fuels (CERCERs and CERMETs, as backup) will likewise need to be considered with a high fraction of the actinide compounds. Coating type, inert matrix material type, and particle type must be evaluated for this type of fuel. Adaptation of classical fast reactor pin-type fuel will likely require the use of advanced cladding materials. Parameters for optimization include pin diameter, plenum size, internal pressurization, clad-pellet gap, and actinide compound type (carbide, nitride, oxide).

The purpose of this task is to design fuel types with the following characteristics:

- High content of heavy atoms in the volume dedicated to fuel
- Use of low absorber/moderator refractory materials
- Geometries allowing for an efficient cooling (pressure drop in the core, thermal exchange coefficient, normal, incidental)
- High level of fission product confinement (direct cycle)
- Resistance to impurities
- Plutonium content in the range of 15 to 20%, ability to incorporate minor actinides in dilution
- Fission rates starting from 5% FIMA
- Consideration of a strong interaction reprocessing/manufacturing (integrated onsite cycle).

The first step in the fuel development process will be to define criteria for fuel selection based on core neutronics, thermal design, and compatibility with an integrated fuel cycle. This is discussed in detail in Section 3.4.1.4. Fuel design can then begin on fuel concepts that meet the selection criteria. The first fuel design calculations conducted on selected fuels will rely on existing or extrapolated data and on current software. Models are improved when new data are obtained (see below). The following points are to be addressed first:

- Extrapolation of particle (PMR type, BISO, TRISO) concepts in fast fluence and feedback on design and materials, and use of dense actinide compound (carbide, nitride)

- Composite fuels: CERCERs, CERMETs, and high fraction of actinide compounds. Optimization of kernel size and shape, management of fission gas release, feedback on materials, and use of dense actinide compound (carbide, nitride)

- Comparison of particle type and CERCER type, and interest of intermediate design (coated kernels in inert matrix)

- Pin type fuel, optimization (e.g., diameter, plenum, pressurization, clad-pellet gap), dependence on the nature of actinide compound (carbide, nitride, oxide), and feedback on the cladding material (specification).

3.4.1.2 Basic Data and Research Concerning Inert Materials and Actinide Compounds.

There is a need for fundamental materials property data for the inert materials and actinide compounds that will be considered for the fuel form concepts developed as a result of this plan. In essence, thermal, mechanical (strength, ductility), and transport properties associated with fission product diffusion and permeation will need to be obtained. In addition, the interaction of materials with the ambient reactor atmosphere and impurities will require quantification, as will the chemical reaction behavior between the inert materials and the actinides. This data will eventually be required for both as-manufactured and irradiated conditions. Depending on the results of the fuel concept down-selection process, the fuel could be in bulk form, such as pellets, sticks, wires or foam, or particles imbedded in a matrix. The fuel could be coated or uncoated. Material parameters that could affect properties could be stoichiometry with respect to oxygen, carbon, or nitrogen, impurity concentration or doping elements, alloy composition, microstructure, and porosity.

Experimental approaches to obtaining this data start with small sample thermochemical experiments, progress to high-temperature heavy ion irradiation, and graduate to irradiation in a fast flux facility such as Phenix. Irradiation of prototype fuels in a materials test reactor will also need to be performed. The sequence of irradiations will begin with uranium only, then to uranium/plutonium, and finally to uranium/plutonium/minor actinide-bearing fuels.

A tentative list of materials for consideration under this task is as follows:

Inerts

- SiC-α,β, ZrC, TiC
- ZrN, TiN, AlN, Si_3N_4
- MgO, $ZrYO_2$, CaO, Y_2O_3
- Cr, Zr ,V, and intermetallics.

In this list, SiC grades are of particular interest.

Actinides

- UPuC

- UPuN.

Recommended tasks to be completed under this heading include:

- Specification and improvements with the aim at obtaining better behavior:
  - As-manufactured (different methods, see below)
  - When irradiated.
- Exploration of physical forms of interest (candidates for participating in the design of fuels) are:
  - Massive forms (pellets, matrix, sticks, foams)
  - Thick coatings and thick coatings reinforced with fibers
  - Thin coatings
  - Thin layers
  - Wires
  - Porous layers.
- Measurement of properties of concern:
  - Thermal properties
  - Mechanical properties (strength, ductility)
  - Transport – diffusion, permeation of fission products
  - Interaction with environment including pollutions by helium and chemical interaction between inert and actinides.
- Parameters that could be manipulated:
  - Stoichiometry
  - Impurities and introduction of doping elements
  - Alloys
  - Microstructure and tailored porosity.
- Experimental techniques involved:
  - Thermochemistry on small quantities
  - Heavy ions high-temperature irradiation
  - Fast neutron irradiation (Phenix).

These data will be gathered in an open database and used to feed existing calculational codes already using explicit description of fuels or homogenization methods. Later, irradiation of prototype fuels in an MTR reactor will be undertaken to verify codes and expected fuel performance. They will include: first, enriched uranium; then, uranium and plutonium; and finally, uranium, plutonium, and minor actinides.

3.4.1.3 Innovative Techniques for Fuel Manufacturing (considering reprocessing technologies—see Section 3.4.2).

The integrated onsite recycling of fuel proposed in this R&D plan calls for several innovations in the design and fabrication of the fuel form. Those innovations and the issues relating to them have a close relationship to R&D needs for the processing of GFR fuel (see Section 3.4.2).

The first issue that requires consideration is the development of techniques to fabricate the actinides into the selected fuel form. These fuel forms may require innovative methods, such as vapor deposition or impregnation, which could call for specific developments to monitor such techniques. For dispersion or CERCER (or CERMET) fuels, fabrication techniques will also require development. Pin-type fuels require development of advanced, high-temperature, ceramic-based cladding to confine fission products. All of these potential fabrication techniques must be developed considering the entire fuel cycle, including a possible onsite integration of fuel cycling, including processing for actinide recovery and potential requirement for remote fabrication. Tasks to be performed under this heading include developing:

- Processes for manufacturing actinides in the form of spheres (kernels), wires, or sticks
- Techniques of deposition of actinide compounds and characterization of deposits
- Techniques of deposition of inert materials and characterization of deposits
- Robust techniques for manufacturing CERCERs and CERMETs in relation with reprocessing
- Ceramic clads and thick coatings liable to act as an efficient barrier to fission products.

3.4.1.4 Development Plan.

3.4.1.4.1 2002—

A tradeoff study is first needed to determine which candidate fuels are most suitable for GFR utilization. This will include gathering available thermodynamic and materials property data and assessing behavior under steady-state and loss-of-pressure conditions. Thermodynamic stability, heat transfer properties, and mechanical behavior will be the primary considerations. This assessment will include the analysis of the suitability of mixed actinide carbide and nitride fuel types, the suitability of inert matrix materials for use with actinide-bearing fuel particles, the compatibility of these fuels with ceramic and metal matrix materials and cladding, and the need for and types of particle coatings required. Possibilities for composite fuels, more traditional coated particle gas-reactor fuels, and pin-type fuels will be considered. This effort must be closely coordinated with core design and fuel recycle efforts to ensure that fuel concepts are selected that meet core physics and safety requirements, recycle requirements, and a core design that takes into account fuel limitations. Where properties data are not known and deemed to be critical to fuel design, the thermal and mechanical properties of candidate inert materials will be measured at temperatures up to 1600°C. One primary and two backup fuels should be selected for further investigation based on the combination of screening studies. The possibility for irradiation of fuels in a fast-spectrum in such facilities as BOR-60 and JOYO after the scheduled Phenix shutdown should also be investigated.

Tasks to be initiated first include:

- Selection and optimization of the design of candidate fuels based on three different technologies and using literature data for materials:
  - Particles
  - Composite CERCER
  - Ceramic clad pin type.

- Definition and manufacturing of samples of inert materials for screening tests
- Characterization of manufactured materials
- Starting screening tests, including heavy ion irradiation
- Preparation of a preliminary project of fast neutron irradiation (e.g., Phenix):
  - Of candidate inert materials
  - Of PMR particles.
- Comparison of carbides and nitride as candidate actinide compounds
- Assessment of chemical interaction risks in between actinides and inert candidate materials (assuming thermodynamic equilibrium)
- Selection of candidate manufacturing processes in relation to reprocessing concerns
- Examination of irradiation capabilities at JOYO or BOR 60 and assessment of a gas loop in MTR reactors (Osiris, HFR).

3.4.1.4.2 2003–2004—The work scope for 2003–2004 will include the following items:

- Basic research in material science
- Characterization of materials after heavy ion irradiation
- Irradiation with heavy ions at high temperature
- Work on microstructure liable to enhance behavior when irradiated
- Preparation of the irradiation of inerts and particles, if available (e.g., in Phenix)
- Experiments on chemical interaction involving small quantities of various actinides
- Progress on modeling using fresh data
- Tests of key techniques for manufacturing involving inerts and actinide simulants first, and natural uranium compound next
- Equipment of alpha lab with such techniques
- Definition of irradiation devices in test reactor(s).

3.4.1.4.3 End of 2004—The work scope for the end of 2004 will include the following items:

- Material irradiation in Phenix ready to be loaded in the reactor
- Confirmation for the choice of a reference design of fuel for GFR and two alternatives
- Preparation of a preliminary project of irradiation of prototype fuels based on uranium-enriched fissile material, and representative of the main technologies contributing to the choice above.

Gas-cooled fast reactor development partners will, in conjunction with existing programs, begin to develop conceptual designs of fuel fabrication/refabrication facilities for GFR fuels.

Thermal stability tests between unirradiated depleted uranium fuel and candidate inert materials need to be conducted at temperatures up to the anticipated maximum fuel temperature on loss-of-flow. Thermal cycling tests of fuels need to be conducted, as required, for composite fuel designs selected for further development. Mechanical properties will be determined after thermal cycling to see if matrix damage will occur. The thermal conductivity of fuels need to be measured before and after thermal cycling to determine the effects of matrix cracking, if present. The mechanical durability of candidate inert coatings will be examined at various temperatures.

Fuel specimens containing uranium need to be prepared for irradiation and inserted into MTRs in the GFR-I tests. This irradiation test should occur under projected nominal, steady-state GFR operating conditions. Although the limitations of testing in a thermal spectrum are understood, the MTRs offer opportunities for cost-effective early fuel screening tests. The specimens should be irradiated in a facility that allows feedback control of fuel temperature and incorporates online fission gas monitoring. Material test specimens should also be included in these tests.

Measurement of thermochemical, thermophysical, and materials property data should continue on reference and backup fuels. The development of fuel performance models and a fuel performance code framework will be necessary.

Fabrication of plutonium and minor actinide-bearing specimens should begin based on the results of uranium fabrication and thermal compatibility tests. These specimens should be subject to out-of-pile thermal compatibility tests.

The choice of primary and backup fuels should be reassessed based on the combination of ANL and CEA out-of-pile studies to date. Development of a fast-spectrum test plan and test design focused on the best fuel candidates should continue.

3.4.1.4.4 2005–2006—The work scope for 2005-2006 will include the following items:

- Basic research in material sciences (including modeling) continuing
- Research on U-Pu fuels
- Irradiations in Phenix.

3.4.1.4.5 End of 2006—The work scope for the end of 2006 will address realization of irradiation devices and manufacturing on uranium-based prototype fuels ready for irradiation.

Standard post-irradiation examination of uranium-bearing GFR fuels irradiated in MTRs (GFR-I tests) should be completed in this time frame, including dimensional measurement, density measurement, radiography, gamma scanning, and metallography. The high-temperature mechanical properties of candidate fuels and inert materials that were irradiated in MTRs will be measured in the hot cell facilities.

Furnace testing of GFR-I fuels irradiated in the MTRs needs to begin. These tests will heat irradiated fuel and measure fission gas release as a function of temperature up to 1600°C. These tests are applicable to all fuel types. Chemical interaction tests should be conducted for fuel and inert materials irradiated in MTRs and compared with out-of-pile results.

Fuel specimens containing plutonium and minor actinides should be prepared for irradiation and irradiated in MTRs (GFR-II tests) under nominal, steady-state conditions based on results of the uranium-fueled GFR-I experiment. The specimens should again be irradiated in a facility that allows feedback control of fuel temperature and incorporates online fission gas monitoring. Material test specimens should also be included in this test.

Materials irradiated in the GFR-I test should undergo post-irradiation examination, including dimensional and density measurement, x-ray diffraction, metallographic examination, and materials property testing. Fuel performance code development should continue and incorporate GFR-I irradiation test data and advances in material property data. The choice of primary and backup fuels should be reassessed based on the combination of in-reactor and ex-reactor studies to date, and by agreement between the two concerns. Measurement of thermochemical, thermophysical, and materials property data will continue on reference and backup fuels.

Development of the fast-spectrum GFR-III irradiation test and test vehicle should continue, taking into account the candidate facilities offering fast neutron irradiation conditions and possible dedicated projects of experimental reactors for fully representative testing conditions.

3.4.1.4.6 2007–2010—The work scope for 2007 to 2010 will include the following items:

- PIE of experiments in Phenix
- Continuing basic research in material sciences (including Modeling)
- Research on uranium, plutonium, and minor actinide-bearing fuels
- Irradiation of prototype uranium-based fuels.

Standard post-irradiation examination of plutonium and minor-actinide bearing GFR fuels irradiated in the MTRs (GFR-II test) should be completed in this time frame, including dimensional measurement, density measurement, radiography, gamma scanning, and metallography. The high-temperature, mechanical properties of candidate fuels and inert materials that were irradiated in MTRs should be measured in hot cell facilities. Differences in the irradiation behavior of plutonium and uranium fuels need to be assessed.

Materials irradiated in the GFR-II tests should undergo post-irradiation examination including dimensional and density measurement, x-ray diffraction, metallographic examination, and materials property testing.

Furnace testing of GFR-II fuels irradiated in MTRs should begin. These tests will heat irradiated fuel and measure fission gas release as a function of temperature up to 1600°C. These tests are applicable to all fuel types.

Fuel performance code development should continue and incorporate GFR-II irradiation test data and advances in material property data. The choice of primary and backup fuels needs to be reassessed based on the combination of in-reactor and ex-reactor studies to date. Measurement of thermochemical, thermophysical, and materials property data will continue on reference and backup fuels, and the need for further thermal spectrum reactor testing will be assessed.

Fuel specimens containing plutonium and minor actinides will be prepared for irradiation and irradiated in a fast spectrum irradiation test (GFR-III test) under nominal, steady-state conditions based on results of the thermal spectrum experiments. The GFR-III fast-spectrum irradiation test vehicle will be fabricated and inserted into a reactor.

3.4.1.4.7 2011–2015—The work scope for 2011-2015 will include (1) PIE of uranium-based fuels, and (2) irradiation of uranium, plutonium, and minor actinide-bearing fuels. The GFR-III fast spectrum irradiation test of fuels containing plutonium and minor actinides should be fabricated and submitted to irradiation under near-prototypic conditions. Fuel designs selected should be based on the results of the GFR-II tests, fuel performance modeling, materials irradiations, and out-of-pile data. A series of fast-spectrum tests will be conducted in available fast neutron irradiation facilities, including dedicated projects of experimental reactors for fully representative testing conditions. The GFR-III fast spectrum fuel irradiation tests should undergo post-irradiation examination during this time.

3.4.1.5 *Cost Elements.*

- *2002–2006*—20 FTE/year plus \$7M
- *2007–2015*—12 FTE/year plus \$4M.

3.4.2 Gas-cooled Fast Reactor Fuel Processing

Fuel cycle technologies need to be further developed or adapted to allow for the recycling of actinides while preserving and reinforcing the economic competitiveness of the nuclear option in the medium and long term. The potential advantages of hydrometallurgical and pyrochemical processes for the different types of fuels considered—including actinide fuels—need to be assessed with a view to selecting the most appropriate options for the next generation fuel treatment techniques and taking into account the advantages of onsite treatment techniques for minimizing the transport of nuclear materials and enhancing proliferation resistance.

The objective is to seek solutions that: (1) minimize the release of effluents to the environment (gaseous and liquid waste); (2) take into account, starting at the design stage, the management of induced secondary waste from treatment and conditioning; (3) simplify the integration of treatment and fuel manufacturing operations; (4) allow for integrated in situ treatment; and (5) develop the capability to treat PMR particles as well as CERCER fuels (or CERMET as backup). Technologies able to separate inert materials (matrix, coating) from actinide compounds will be looked for.

An important phase of the R&D program will be to perform an experiment on the selected fuel concepts to demonstrate at a significant level (few kg of fuel) the treatment of irradiated fuel and the possibility of its re-fabrication. This means that the objective is to have selected and demonstrated the scientific viability of a process by the end of 2011. After process screening, mostly with surrogate materials (2002–2007), more in-depth studies will be performed in hot laboratories (2008–2011) at small scale with the selected treatment process, using irradiated fuel samples provided by the irradiation program for fuel development. The final phase of the development program will consist of demonstrating the technologies associated with the fuel cycle processes in a reduced-scale pilot plant with surrogate materials (2012–2015) before constructing the fuel cycle plant for the GFR prototype system.

3.4.2.1 **General Scope of Development Needs.** GFR fuel concepts under consideration include carbide, nitride or oxide-coated fuel particles, or dispersions of fuel particles in a ceramic matrix or metal as a back up. Accordingly, the ceramic matrix could be a carbide, a nitride, or an oxide. There is virtually no experience in processing coated particle or inert matrix dispersion fuels, and a significant effort will be required to develop the separations technologies for such fuels. Primary among the problems that must be solved is the separation of the coatings or inert matrix from the fuel particles in a way that minimizes the generation of high-level waste. Both aqueous and pyrochemical processing methods can be applied to these inert-matrix fuels, and studies to evaluate processing concepts would be the first step in the development program. Candidate processes with reasonable expectations of technical feasibility will be compared in detail at the conceptual stage, to include evaluations of mass-balance flowsheets and estimates of equipment and facility requirements necessary to meet established criteria for product quality and throughput capacity.

The GFR concept assumes a closed fuel cycle with the recycling of fissile and fertile actinides. The implementation of an aqueous processing system to deal with inert matrix fuel would require the development of optimized fuel dissolution procedures. In case of nitride ceramics, the processing techniques should account for the possibility of a need for recovering enriched nitrogen to avoid extensive ^{14}C production. Existing methods for extraction of uranium, plutonium, and thorium can be applied, but advanced methods for dealing with the minor actinides would necessarily require validation in the presence of the inert matrix materials. A strong improvement of the GFR fuel cycle would be the possibility to recover the actinides all together without any intragroup separation, not only because it would simplify the overall process scheme but also because it would greatly enhance the proliferation resistance feature of the fuel cycle.

Pyrochemical methods developed in concept for accelerator-driven fast spectrum transmutation systems are directly applicable for GFR inert matrix fuel processing. Conceptual processes for treatment of carbide, nitride, or oxide dispersion fuels in ceramic or metal matrices have been evaluated and appear technically feasible. However, extensive experimental work is required so that the process concepts can be proven feasible for fuel treatment at production scale.

It is highly likely that hybrid processes will prove to be technically and economically superior in the long run, and efforts will be extended to evaluate the optimum combinations of aqueous and pyrochemical process steps. For example, a pyrochemical step may be preferred for the digestion of the inert matrix material, with an aqueous step used for the separation of some or all of the actinides from the fission products.

One important objective of the GFR fuel cycle is to design onsite integrated processing. Among others, this feature means, for both pyrochemical and hydrometallurgical methods:

- Integrating fuel reprocessing and re-fabrication within the same facility (the “all-in-one” concept). Associated with the recycling of all the actinides, including the minor ones, this implies innovative processing methods from both the chemistry and the engineering point of view.

- Minimizing gaseous release to the environment to cope with the level of release of the reactor itself. Thus, high performance trapping and conditioning technologies for volatile radioactive isotopes (^{3}H, ^{14}C, ^{85}Kr, ^{129}I) must be designed

- Having compact technologies at all stages of the process, even for ancillary operations such as the vitrification unit.

3.4.2.2 **Approach to be Followed.** Due to the wide diversity of GFR fuel types under consideration (i.e., coated particle fuels and inert matrix CERCERs and CERMETs, with a variety of actinide contents), it is proposed to build the process development around three concepts:

- *Recycling Requirements and Specifications*—Set of functions needed to perform actinides recycling. The requirements and specifications define performance levels to be reached and constraints to be followed, in particular, for onsite processing.

- *Technological Block*—Unit operation totally or partially performing one of the above functions and that could be applied to several types of fuel. Actinides group separation by hydrometallurgy or fuel electro-dissolution in a molten salt are examples of technological blocks. One block can be made of several “sub-blocks.”

- *Conceptual Process*—Coherent association of technological blocks meeting the recycling requirements and specifications.

Development of conceptual processes for the reprocessing of all of the current candidate GFR fuel types will be carried out by evaluating process feasibility (including bench scale test on surrogate material whenever it is necessary) of technological blocks, completing mass balance flowsheets, and preparing estimates of facility requirements together with preliminary operating and capital cost estimates. The purpose of this activity is to screen out those technological blocks that have little chance for success, without the expenditure of a great deal of time and money. Experimental validation of promising process concepts will follow this screening process, with close coordination between the reprocessing experts and fuels experts so that appropriate emphasis can be placed on the most promising fuel types and processing methods. The scale of experimentation will purposely be kept small until the evaluation of candidate fuels and processes has reached the stage that each key elementary step (dispersion, separation, re-fabrication) has been successfully tested with genuine or surrogate material (depending upon availability). That stage characterizes the scientific feasibility of the conceptual process, and, when it is reached, commitments to larger-scale process development will be appropriate.

Initial experiments for process validation will be carried out with simulated fuel materials in the case of the CERCER and CERMET fuel types, and with unirradiated TRISO fuel in the case of coated-particle fuel. The purpose of these experiments will be to confirm the technical feasibility of the candidate processes and to identify process steps that must be altered for improved performance. Some of the testing may be limited to key process steps, such as removing and disposing of the carbonaceous coating materials from TRISO fuel. When needed, the experimentation can shift to work with irradiated fuels, if they are available, once the post-irradiation examination requirements have been met. It will be necessary to assure a sufficient quantity of fuel in early scoping irradiations to meet the needs of the chemical separations process development activities. Testing with irradiated fuels will be particularly important in providing an early indication of any problems attributable to specific actinide or fission product elements.

Once the initial process experimentation is completed, it will be time to choose a reference process and fuel (together with back-up options) so that process scale-up can begin. This is the point at which the development program enters the realm of process engineering, with the development of process equipment designs intended for operation at throughput rates characteristic of commercial applications. The integration of unit operations and materials handling is an important element of this engineering process. It is anticipated that certain elements of process equipment can be tested at a fraction of the size expected for production-scale equipment, while others may require design validation at near-prototypic size. The process-engineering phase will begin with tests of individual process stages, then move to an integrated test of the full flowsheet for selected processes. That stage, where all the elementary steps have been successfully tested in integration and using devices prototypic of industrial applications, will demonstrate the "technological feasibility" of the recycling scheme.

Construction and operation of a fully integrated prototype process line will be the final step in the development program. Timing for this pilot-scale facility must be consistent with the GFR prototype.

3.4.2.3 **Development Plan.** The proposed development plan consists of five distinct phases extending over a period of nearly 20 years.

3.4.2.3.1 Phase 1, 2003–2004—Selection of Conceptual Processes—

- Estimation of onsite production-scale facility requirements and specifications
- Development of conceptual process diagrams
- Assessment of feasibility of technological blocks
- Development of mass balance flowsheets for promising concepts
- Performance of feature tests of key technological blocks for selected process concepts
- Evaluation of processability of candidate fuel types.

3.4.2.3.2 Phase 2, 2005–2008—Development of Selected Processes and Preparation for Small-Scale Hot Test—

- Basic research in chemistry and chemical engineering
- Optimization and experimental demonstration of selected technological blocks with simulated, unirradiated, surrogate fuels
- Preliminary evaluation of commercial feasibility of process concepts
- Specific equipment design and tests
- Preliminary assessment of needed infrastructure for small-scale hot test
- Start to build or adapt infrastructure.

3.4.2.3.3 Phase 3, 2009–2012—Laboratory-Scale Hot Testing and Process Scale-up—

- Finish build or adapt infrastructure
- Final design, construction, and implementation of equipment for processing of irradiated fuels at laboratory scale
- Safety assessment on the chosen technologies
- Advanced preliminary design for pilot-scale facility
- Complete site implementation studies for pilot-scale facility
- Launch administrative authorization procedure
- Start mockup design, scale-up, and cold testing of specific devices
- Experimental laboratory-scale demonstration of selected process concepts with irradiated fuels.

3.4.2.3.4 Phase 4, 2013–2020—Pilot-Scale Design and Construction—

- Basic and detailed design of the pilot-scale facility
- Equipment design and tests
- Procurement for buildings and equipment
- Safety analysis and environmental impact study
- Construction and cold-test.

3.4.2.3.5 Phase 5, 2021–2025—Pilot-Scale Operation—

- Tests with un-irradiated fuels
- Processing of irradiated fuels (coming from GFR prototype).

3.4.2.4 Estimated Costs.

- *Phase 1*—17 FTEs per year + $4 million for materials and services
- *Phase 2*—17 FTEs per year + $18 million for materials and services
- *Phase 3*—20 FTEs per year + $28 million for materials and services

- *Phase 4*—26 FTEs per year + $24 million for materials and services.

3.4.2.5 Major Milestones.

- *April 2003* Edit onsite production-scale facility requirements and specifications
- *December 2004* Complete initial screening of candidate technological blocks
- *December 2006* Evaluate commercial feasibility of the selected reference process concept
- *December 2008* Complete cold demonstration tests of key technologies
- *December 2009* Complete flow sheet for small-scale hot testing
- *November 2010* Complete design of prototype equipment for small-scale hot testing
- *December 2010* Procurements for small-scale hot testing equipment
- *December 2012* Complete laboratory-scale hot testing of selected process concepts
- *July 2013* Final preliminary design of the pilot-scale facility
- *December 2014* Administrative authorization to build the pilot-scale facility
- *July 2015* Complete design of prototype equipment for pilot-scale hot testing
- *December 2015* Safety analysis report and environmental impact statement approval
- *December 2016* Final manufacturers procurements edition
- *December 2019* Authorization to put the pilot scale facility in active operation
- *April 2020* Start pilot-scale processing with simulated fuel
- *March 2023* Start pilot-scale processing of irradiated fuels
- *December 2025* Report on the industrial feasibility of the onsite GFR fuel recycling.

3.4.3 Reactor Systems

***3.4.3.1* High-Temperature Materials.** The study of high-temperature materials and the technology of the helium circuits have a common synergetic basis for the development of high-efficiency power conversion. The basic research and modeling efforts necessary to support the innovations needed for the conceptual designs also appear as a common basis. Analyses of the advantages of various refractory materials and alloys should be complemented by basic research studies. A preliminary selection of materials is foreseen by 2006 for:

- A reference set of refractory materials (ceramics and metals)
- Advanced ferritic and martensitic steels (to be tested in, for example, Phenix, JOYO, or BOR-60)
- Advanced austenitic steels and oxide-dispersed steels (to be tested in, for example, Phenix)
- In-service behavior of steels and superalloys.

As far as specificities of GFR are considered, the main challenge is the in-vessel structural materials, both in-core and out-of-core, that will have to withstand fast neutron damages and temperatures up to 1600°C in-core, when considering accidental situations. Ceramic materials will, therefore, be the reference option for in-core materials, and composite CERMET structures or intermetallic compounds will also be considered as a back up. For out-of-core structures, metal alloys will be the reference option.

The corresponding R&D program is closely linked to the fuel development program with a screening phase for material irradiation and characterization between 2002 and 2005, selection of a reference set of materials for core structural materials in 2006, and optimization and qualification under irradiation from 2006 to 2012. The objective is to be in a position to fabricate a prototype core in the 2012–2015 timeframe.

3.4.3.1.1 General Approach—In concept, GFRs with helium as the primary coolant that also drives the turbine (direct cycle power conversion) should be able to achieve high core outlet coolant temperatures (~850°C) and, therefore, high plant net thermal cycle efficiency. Maintenance, repair, and in-service inspection of the GFR should be relatively simple, and there is the potential for low life-cycle maintenance costs.

This promise, however, requires the choice of structural materials that can survive the fast neutron fluence goals and handle the high operation temperatures over the design life. Accident temperatures in the core could even be as high as ~1600°C or more.

In addition, GFRs also have the potential to be more than just producers of electric power. Their high-temperature potential allows them to be multifunction facilities and potential sources of hydrogen fuel for distributed needs, and process heat for high-grade industrial consumption. This requires the use of high-temperature materials.

Regarding GFR core structural material, ceramics materials (CER) and composite materials with ceramic matrix (CERCER, CER_fCER, CERMET) present the greatest promise and will be evaluated for use in a high-temperature nuclear environment ranging from 500°C to about 1000–1200°C in normal operating conditions, and up to 1600–1800°C in accidental transients. As a backup plan, refractory alloys that have been examined by the fusion energy program for operation at high temperature, such as T-111 (Ta-8W-2Hf) or Mo-0.5Ti-0.1Zr, will also be evaluated.

3.4.3.1.2 Ceramic Materials for GFR Applications as Core Structures—The program consists of designing, fabricating, and characterizing nuclear ceramic materials (CER) for use in the core of G-CFR as a large range of products like monolithic CER, CER coating or CER/CER, CER_f/CER, and CER/MET composites. The essential goal of this program is to select, for each component, the most promising candidate exhibiting the best compromise between the following key properties:

- Fabricability and welding capability
- Physical, neutronic, thermal, tensile, creep, fatigue, and toughness properties: initial characteristics and assessment of their degradation under high neutron flux and dose
- Microstructure and phase stability under irradiation
- Irradiation creep, in-pile creep, and swelling properties
- Initial and in-pile compatibility with helium (and impurities) and actinide compound.

The main core applications aimed at here are the following inert structures involved in the different fuel concepts:

- Particle coatings and basket containing the particles for this concept
- Inert matrix, casing, and gas tubing for the composite fuel concept
- Clad for the solid solution fuel concept.

Efforts will be made preferentially on the most promising ceramic solutions to be chosen among carbides (preferred option), such as SiC, ZrC, TiC, and NbC, or other materials like nitrides (TiN, ZrN) and oxides (MgO, Zr(Y)O_2). The effort will also include intermetallic compounds like Zr_3Si_2 as a promising candidate of fast neutron reflector. Limited work on Zr, V, or Cr as the metallic part of the back-up CERMET option will also be carried out.

The objectives are twofold:

1. In the preliminary period (~ 2002-2010), to gain, on samples or on first prototypes, sufficient basic knowledge on the different prime candidates (up to 2006) and to validate initial choices or to propose alternative materials (up to 2010) via experimental neutron irradiations of materials.

2. In the demonstration period (~ 2011-2016), to validate fabrication processes and in-pile behavior of actual components, like fuel pin, basket, tubing, or casing.

3.4.3.1.2.1 Preliminary Period.

- *Task 1, 2002-2006*—Selection, design, and fabrication of a first set of selected carbide ceramics (in a first time, focusing on SiC, ZrC with parallel backup studies on high-temperature refractory materials) for preliminary supplying of:
  - Samples for basic studies, mechanical and physiochemical characterizations, or samples for material irradiations and post-irradiation (or in-pile) examinations: CER thin layers or CER, CER/CER, and CER_f/CER thick plates
  - Prototype of final object, like tube for fabrication, demonstration, and characterizations: CER, CER/CER, CERf/CER, and CER/MET
  - CER being conventional or nanostructured monolithic ceramic to improve mechanical behavior of the material
  - CER/CER being a composite formed by a first CER(1) finely dispersed in the second CER(2) that constitutes the matrix to take advantage of the good but different properties (for example, thermal properties for one side and mechanical properties for the other side) of both the ceramics (1 and 2)
  - CERf/CER being a matrix (SiC or ZrC) reinforced by fiber (SiC) with or without C at the interface to optimize initial toughness and mechanical behavior under irradiation
  - CER/MET being a composite formed by alternated layers of ZrC (or SiC) and near-refractory metal to take advantage of the good but different thermal and mechanical properties of metal and ceramic in the entire operating temperature range.
- *Task 2, 2002–2006*—Out-of-pile characterization of basic and mechanical properties at low and high temperatures. Preliminary studies on samples and tubes:
  - *2002–2004*—Feasibility studies of dedicated mechanical tests on thin layers, thin plates, and tubes with the constraint that these tests also be feasible in hot cells to compare irradiated and unirradiated state
  - *2002–2006*—Experimental determination of the mechanical properties
  - *2003–2006*—Understanding studies based on microstructural examinations to identify the mechanisms of failure

- *2003–2006*—Modeling the mechanical behavior focusing on creep and toughness properties and on mechanisms of failure of composites
- *2002–2006*—Simulation of the irradiation effects by charged particles, microstructural examinations, development of models describing and anticipating the irradiation-induced degradation of the thermal and mechanical properties
- *2002–2006*—Corrosion studies; building a dedicated experimental device to study corrosion phenomena at 1000, then 1600°C; conducting tests; and understanding work
- *2006*—Preliminary report R1: proposal of ceramic materials for GFR core applications.

* *Task 3, 2002–2010*—First validation task that consists of testing the most promising candidates and selecting prime candidate materials for component fabrication via the implementation of the following neutron irradiations and related PIE:
  - Experimental irradiation in Phénix to reach high dose (50–60 dpa) in the lower range of operating temperatures (500–750°C). Here an important first screening test on different materials and post-irradiation properties will be realized because the material experimental rig of Phénix is capable of containing a large number of samples.
  - Experimental irradiation in Osiris to reach high temperature (1000°C) at low (~10 dpa) but sufficient dose to anticipate phenomena occurring under irradiation at high temperature. This experiment is dedicated to the behavior characterization of a limited number of most promising materials.
  - PIE in Saclay's hot cells—nondestructive and mechanical testing (mainly on subsized samples studied in the out-pile work), and microstructural examinations.
  - Experimental irradiation in the Advanced Test Reactor to reach high temperature (1000°C) at low (~10 dpa) but sufficient dose to anticipate phenomena occurring under irradiation at high temperature. This experiment is dedicated to the behavior characterization of a limited number of most promising materials.
  - PIE in Argonne National Laboratory's hot cells—nondestructive and mechanical testing (mainly on subsized samples studied in the out-pile work), and microstructural examinations.
  - Preliminary dossier of feasibility (R2) of ceramic materials for G-CFR core applications.

3.4.3.1.2.2 Final Demonstration Phase (2007–2016).

* *Task 4, 2007–2009*—Continuation of the preliminary out-of-pile work to qualify the reference materials chosen for the core structures (fuel and reflector subassemblies):
  - Experimental work about thermo-mechanical properties, corrosion resistance, and simulation of the irradiation effects
  - Modeling work to give a physical basis to the design rules.

* *Task 5, 2007–2010*—Mechanical analysis of core structures, rules for lifetime prediction, codes, and standards:
  - Analysis of methods allowing guarantee of the structural integrity of ceramic core structures, thus taking into account the special occurrence of highly irradiated materials exhibiting low ductility and low toughness properties
  - Compiling design rules for codes and standards applied to GFR.

- *Task 6, 2007–2016*—Final validation via the fabrication of the different structures (pin, tube, basket, casing, reflectors, etc.) of an actual subassembly that will be irradiated in realistic conditions of a helium-cooled reactor prototype:
  - Specifications of materials, final objects, welding, and inspection
  - Fabrication of the components
  - Implementation of the demo irradiation and related PIE
  - Final report R3.

3.4.3.1.2.3 R&D Schedule, Personnel and Expenditure. The duration of all the tasks and subtasks are presented in Table 18, as are a best estimate of the needs in:

- Personnel by year (in the bottom row)
- Total expenditure, including personnel cost (in the right-most column).

For the backup plan on refractory alloys, the estimate is given below:

Task 1	$3M
Task 2	$6.5M
Task 3	$2.2M Irradiation (single instrumented irradiation campaign in ATR)
	$2.2M PIE
Total	**$13.9M million**

3.4.3.1.3 R&D Program on Materials for GFR Applications as Out-of-Core Structures—The program consists of designing, fabricating, and characterizing materials for use in out-of-core components in GFR. The essential goal of this program is to select for each component the most promising candidate exhibiting the best compromise between the following key properties:

- Fabricability and welding capability
- Physical, neutronic, thermal, tensile, creep, fatigue and toughness properties—initial characteristics and assessment of their degradation under low to moderate neutron flux and dose
- Microstructure and phase stability under irradiation
- Irradiation creep, in-pile creep, and swelling properties
- Initial & in-pile compatibility with helium (and impurities).

The core applications aimed at here are as follows:

- Internal structures, mainly the upper and lower structures, shielding, the core barrel and grid plate, gas duct shell, and the hot gas duct. The candidate materials involved are coated or noncoated ferritic-martensitic steels (or austenitics as alternative solution), other Fe-Ni-Cr bases (Inco 800), and Ni-bases alloys.
- Pressure vessels (reactor and energy conversion) and cross-vessel—the main candidates are 21/4 Cr and 9-12 Cr martensitic steels.

Table 18. GFR R&D schedule and costs.

Year R&D Task		2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	(k$)
1	**Materials**																
1.1	Designing																266
1.2	Samples fabr.																971
1.3	Prototype fabr.																1,875
2																	
2.1	Feasibility studies																465
2.2	Mechanical properties																1,396
2.3	Microstructural examinations																559
2.4	Modeling																625
2.5	Simulation studies																1,862
2.6	Corrosion studies																1,370
2.7	*R1:* Material proposal																266
3	**First validation**																
3.1	Low temp. irradiation																1,955
3.2	High temp. irradiation																2,354
3.3	PIE																2,660
3.4	*R2:* Feasibility																266
4	**Further studies**																
4.1	Experimental																1,197
4.2	Basic studies																598
5																	
5.1	Design rules																731
5.2	Codification																864
6	Final validation																
6.1	Specifications																798
6.2	Fabrications																3,324
6.3	Demo irradiation and PIE																5,452
6.4	*R3:* Demo																266
	Total (FTE/year)	5	7.5	11.5	12	15	15	15	15	13	7	4	4	4	5	5	30,000

3.4.3.1.3.1 Task 1, 2003–2004—Selection, Design and Fabrication of Potential Out-of-Core Materials.

- Preliminary choice of materials for each component
- Design and fabrication of a first set of selected candidates—preliminary supplying of samples
- Fabrication and welding studies, out-of-pile mechanical properties (including helium impurities effects), and characterizations.

3.4.3.1.3.2 Task 2, 2005–2010.

- Irradiation testing of components in fast neutron spectrum in Phenix
- Post-irradiation characterization
- Compiling design rules for codes and standards applied to GFR.

3.4.3.1.3.3 R&D Schedule, Personnel, and Expenditure.

Task 1	$3M		
Task 2	$2M	Irradiation	
		$2M	PIE
		$1M	Design rules, codes, and standards
Total	**$8M**		

3.4.4 Balance of Plant/Energy Products

The operating conditions and needs for energy conversion are expected to be similar to those for the PMR (see discussion in Sections 3.2.6 and 3.2.7 for energy conversion component development and hydrogen production considerations).

3.4.5 Safety

The design parameters for the GFR will ensure very high performance with respect to both sustainability and safety objectives. The level of power density will be higher than in current or foreseen PMRs. This will call for a safety approach relying on intrinsic core properties and the addition of the necessary safety devices and systems. The objective of the R&D plan is first to define and perform in-depth studies on what will be identified as the safety case, study and experimentally demonstrate safety systems and devices, and reduce as far as possible the need for active systems. A comprehensive R&D plan will include transient fuel testing, of both the developmental and confirmatory kind, and the various phenomenological model and code development to provide the basis for the final safety case. Integrated safety experiments simulating the ‘safety case’ of the GFR will have to be prepared at the beginning of the demo phase by 2012. These demonstrations, as well as those needed for earlier versions of modular GCRs (PMR, VHTR), will require experiments in an integral helium loop (~20 MW).

The advanced GFR will have to be designed to overcome the consequences of the use of a high-pressure gas coolant with poor thermal characteristics. The safety consequences of the high coolant pressure are the potential for LOCAs. In addition to the poor heat removal properties of the coolant at low pressure, the GFR safety case may be further complicated if the core has a low thermal inertia. The early GFRs faced difficulties to safely manage LOCAs as they were designed with a high power density to achieve high breeding performances and short doubling times. Design parameters for new proposed GFRs aim at achieving a good balance to ensure very high performance with respect to both sustainability and safety objectives. More generally, the Generation IV goals require excellence in safety and reliability, a very low likelihood and degree of reactor core damage, and elimination of the need for off-site emergency response.

3.4.5.1 Reactivity Control.

The fast-hardened spectrum design of the GFR offers an opportunity for enhanced reactivity feedback through core expansion that, together with a refractory fuel, would offer promising prospects of surviving anticipated transients without scram, without severe core damage. Nevertheless, specific R&D should be devoted to innovative and possibly passive shutdown systems.

3.4.5.2 Decay Heat Removal.

The innovative GFR technologies and design features that require R&D are intended to overcome shortcomings of the past GFR designs, primarily low thermal inertia and poor helium heat removal capability at low pressure. Various passive approaches need to be evaluated for the ultimate removal of decay heat in depressurization events. The conditions to ensure a sufficient back-up pressure and to enhance the reliability of flow initiation are some of the key issues for natural convection, the efficiency of which will have to be evaluated for different fuel types, power densities, and power units. Dedicated systems, like semi-passive heavy gas injectors, need to be evaluated and designed. There is also a need to study the creation of conduction paths and various methods to increase fuel thermal inertia and, more generally, core capability to store heat while maintaining fuel temperature at an acceptable level. For classical designs, such as fuel pins, this may eventually result in plate-type geometry with web-based conduction. In the limit, this could be block fuel elements with a dispersion-type fuel in a heat-conductive matrix with similarities to prismatic fuel elements. Passively conducting particle beds could also be a possibility. Heterogeneous cores with internal reflectors could be considered. The consideration of heat radiation to remove heat from the core may not be neglected. Dedicated inner core heat exchangers could be considered to remove, by convection, heat transferred by radiation or conduction. The design constraints that may limit the new geometries are the need for an acceptable core pressure drop/circulator pumping power and specifications on fluid induced vibration at full power. In past designs, the high, full-flow gas velocities necessary for full-power heat transfer have had consequences for these design parameters. Passive safety includes not only passive safety at decay heat levels but also passive safety at full power conditions, which is largely dominated by reactivity feedback effects. A design constraint on developing new fuel forms will be the acceptability of the resulting safety coefficients and core expansion mechanisms.

3.4.5.3 Development Plan.

A comprehensive R&D plan will include transient fuel testing, of both the developmental and confirmatory kind, and the various phenomenological model and code development to provide the basis for the final safety case. Integrated safety testing will also be part of the plan. Main milestones would be as follows:

- *2004*—Report on GFR technical and safety options. Description and preliminary design of safety systems
- *2007*—Start of experimental programs in acceptably scaled loops to qualify safety systems
- *2010*—Start of transient testing of irradiated samples of GFR fuels in hot laboratories
- *2012*—Integral safety experiment simulating the ‘safety case’ of GFR.

The estimated total cost of such a program is $80 million.

3.4.6 Economics

No economic R&D needs were identified.

3.4.7 Security

No security R&D needs were identified.

3.4.8 Calculation Tools/Major Codes

The advanced GFR design and safety analysis will require major adaptations or evolutions of calculation tools to accommodate the innovative associated features in the field of core design (new fuel and subassembly forms), fuel composition (homogeneous recycling of minor actinides with a robust on-site integrated cycle), safety devices implementation, and the very important need to demonstrate the safe behavior of the whole system under all operational conditions. This will call for adaptation of the neutronics, thermo-aero-mechanics, operation and safety codes. Furthermore, qualification experiments (e.g., critical experiments and subassembly mock-up testing) have to be considered. A Core Melt Exclusion Strategy should also consider degraded core configurations.

3.4.8.1 Neutronics. The specificities of GFRs (materials, subassembly design, preferential direction for neutron leakage [streaming], high temperatures, particular reactivity effects, etc.) will require at least an increase in the number of nuclides to be taken into account in the neutronic libraries with an extended tabulation in temperature. Enhancement of neutronic calculational tools will be needed for S/A heterogeneity and anisotropy and to accurately model control elements and other nonfueled regions. The neutronics code systems will require qualification through comparison of their predictions to measurements conducted in critical facilities, including critical mock-ups of the particular CFR core designs that will be selected.

3.4.8.2 Thermo-Mechanics-Hydraulics. Codes will be required not only to describe the fuel behavior (see Fuel R&D), but also the global behavior of the fuel subassemblies and their configurations in constituting the core. In that area, fuel subassemblies may present innovative configurations, and the question of their thermomechanic and aerodynamic behavior will be a crucial issue. Code qualification will require experimental work involving instrumented subassembly mock-ups to be tested in representative helium flows.

3.4.8.3 Operation/Control. Current codes aimed at describing the operational behavior of the reactor systems will have to be evaluated in their capabilities to describe the GFRs. Depending on the effort to be expended and the necessary flexibility required for concepts, which may exist in various options for a certain time, the opportunity to develop a new code will have to be considered.

3.4.8.4 Safety. One of the major issues will be to model the depressurization event with and without scram, while taking into account all the modes of heat transfer (conduction, convection, radiation, storage, etc.) and the action of the specific safety device, the design and experimental qualification of which is under the scope of the Safety R&D Plan. For degraded situations, the adequate reference codes will have to be adapted to helium.

3.4.8.5 Development Plan. A comprehensive R&D plan will have to include code development and associated qualification programs that are not already included in the other R&D plans. Some important milestones would be as follows:

- *2004*—Detailed report on code development needs in accordance with the GFR preferred technical options
- *2005–2008*—Performance of experiments on critical mock-ups for neutronics codes qualification
- *2005–2008*—Modeling and qualification of fuel subassembly hydro-mechanic and dynamic behavior (mock-ups) in acceptably scaled loops

- *2008*—Availability of a complete code system with a first level of qualification for GFR calculation.

The estimated total cost of such development is $60 million.

4. RECOMMENDATIONS

A summary report of the candidate GCR system concepts that have the potential to fulfill Generation IV goals [Gas-TWG, 2001] was completed in December 2001. In that report, nineteen concepts were aggregated into four concept sets for detailed evaluation, each represented by a reference concept. The four reference concepts that satisfied this screening for potential were a pebble bed modular reactor system (PBR), a prismatic modular reactor system (PMR), a very high-temperature gas-cooled reactor system (VHTR), and a fast neutron spectrum gas-cooled reactor system (GFR). In this report, the Gas-TWG has further developed the evaluation of these candidate concepts and has recommended the critical and associated R&D scope required to support the design and application of similar or related reactor system concepts. The primary focus in these recommendations is the scope necessary to confirm viability of the concepts and investigate important performance-related issues.

These four concepts comprise a progressive group of reactor systems with complementary R&D activities. These concepts rely on a common R&D pathway composed of basic needs for potential near-term concepts (i.e., PMR and PBR) and more ambitious objectives for the advanced concepts with the potential for extended capabilities (i.e., VHTR as a higher-temperature heat source for more efficient electricity generation and alternate fuel production, and GFR to achieve greatly improved sustainability). As depicted in Figure 21, these concepts provide innovative capabilities through the evolutionary development and implementation of gas-cooled reactor systems.

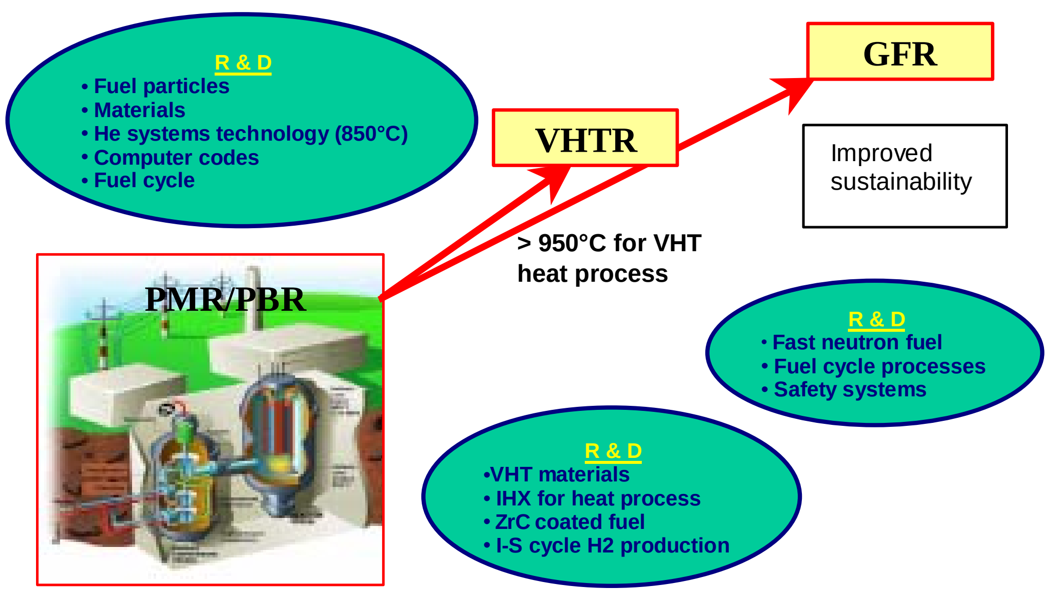


Figure 21. GCR reference concepts provide innovative capabilities through the evolutionary development and implementation of GCR systems.

The following recommendations apply:

- The R&D activities for the PBR and PMR are important precursors for the development of VHTR and GFR. The Technical Roadmap should describe this relationship and provide for periodic review of the status and success of the development and application of these nearer-term concepts to confirm that the ongoing R&D scope is adequately comprehensive. Further, as the practical aspects of funding priorities and cooperative development are realized, particular priority should be those viability or performance issues that are most broadly applicable across the spectrum of GCR systems.

- The VHTR concept should not be approached as a reactor system with a specific operating temperature and particular energy conversion process. Rather, R&D activities should be directed toward achieving the capability for increased temperatures at several points over a range from 950°C to 1100°C, since the materials that may succeed for fuel coating and plant equipment could be expected to change markedly over this range. This approach would not only provide the potential for higher-temperature applications, but also provide materials with additional margins for use at temperatures applicable to PBR and PMR.

  Similarly, for energy conversion development, various candidate applications (e.g., hydrogen production and electrical power generation) and more than one way to achieve an application should be considered (e.g., for hydrogen, steam reforming and thermochemical processes)

- Where available and practical, the R&D activities should be aligned to current and ongoing development activities that are not currently associated with the Generation IV program. For example, the HTTR in Japan provides an opportunity for complementary development. In this case, for instance, an initial research activity could be to review the current scope of HTTR, its status and planned additional scope against the overall R&D activities for a VHTR concept with the intent of reaching an agreement on sharing information and cooperatively using this facility for Generation IV work.

- A particular weakness in the concept evaluations was estimating the expected costs for building and operating the reactor system. This becomes even less certain when attempting to estimate whether and under what conditions the reactor system concept could be expected to be competitive for production of alternate fuels. Economic studies should have an early and generic priority (e.g., to better establish the expected market place for alternate fuels) to address conceptual issues such as the potential economic tradeoffs on small, modular reactor concepts versus the more conventional wisdom regarding large facility economy of scale.

  Further, the utilization of certain of these reactor system capabilities would be expected to be determined not by the marketplace, but by forward-looking governmental policies regarding carbon-based energy resource utilization, nuclear fissile and fertile resource utilization, non-proliferation, and nuclear waste management. R&D activities and studies, the results of which assist in shaping government policies for utilization of these capabilities, should be a high priority in the Generation IV Roadmap scope.

- A weakness in both the concept evaluations and the description of R&D scope regards the approach to addressing proliferation resistance and physical protection. Early studies should be directed at defining the conceptual standards that should be used in characterizing the threats, evaluating the reactor system vulnerabilities and designing reactor systems that have improved capabilities for these considerations. In addition to the obvious desire to ensure the safety of the general population, these standards could be expected to affect conclusions regarding the specific and comparative economic viability of reactor systems.

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